SSI report

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Radiological Protection in Transition

Proceedings of the XIV Regular Meeting of the Nordic Society for Radiation Protection, NSFS
Rättvik, Sweden, 27-31 August 2005

Statens strålskyddsinstitut
Swedish Radiation Protection Authority

SSI's Activity Symbols

Ultraviolet, solar and optical radiation
Ultraviolet radiation from the sun and solariums can result in both long-term and short-term effects. Other types of optical radiation, primarily from lasers, can also be hazardous. SSI provides guidance and information.

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The second largest contribution to the total radiation dose to the Swedish population comes from health care. SSI is working to reduce the radiation dose to employees and patients through its regulations and its inspection activities.

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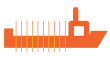

Transport
SSI is involved in work in Sweden and abroad to ensure the safe transportation of radioactive substances used in the health care sector, industrial radiation sources and spent nuclear fuel.

Environment
"A safe radiation environment" is one of the 15 environmental quality objectives that the Swedish parliament has decided must be met in order to achieve an ecologically sustainable development in society. SSI is responsible for ensuring that this objective is reached.

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Emergency preparedness
SSI maintains a round-the-clock emergency response organisation to protect people and the environment from the consequences of nuclear accidents and other radiation-related accidents.

SSI Education
is charged with providing a wide range of education in the field of radiation protection. Its courses are financed by students' fees.

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TITLE / TITEL: Radiological Protection in Transition - Proceedings of the XIV Regular Meeting of the Nordic Society for Radiation Protection, NSFS - Rättvik, Sweden, 27-31 August 2005.

SUMMARY: These proceedings comprise the papers and posters presented at the 14th Regular Meeting of the Nordic Society for Radiation Protection, the theme of which was 'Radiological protection in transformation'. The traditional 'Lindell lecture', presented by H.H. Brunner from Switzerland, highlighted this theme under the title 'From Pandora's Box to Sisyphus'. There were sessions on international developments and stakeholder involvement, on education, training, and measurements, on emergencies, on nuclear installations, on non-ionising radiation, on medical radiation, on industrial uses of radiation, on radiobiology, on natural sources of radiation, on non-nuclear waste, on NKS (Nordic Nuclear Safety Research), on radioecology and artificial radionuclides in the environment, and on regulatory and international activities. In addition to invited lectures and proffered papers, there were educational primer lessons in the mornings and several round-table discussions. In all, there were almost 100 contributions from participants representing at least 10 different countries. The range of different topics covered, the scientific quality of the contributions, and the interest shown in this meeting reflect the high standing of radiological protection in the Nordic countries.

SAMMANFATTNING: Dessa kongresshandlingar är en sammanställning av föredrag och posterutställningar framlagda vid Nordiska Sällskapets för Strålskydd 14:e ordinarie möte under temat "Strålskydd i förändring". Den traditionella "Lindell-föreläsningen" hölls av H.H. Brunner från Schweiz och hade titeln "Från Pandoras Ask till Sisyfos". Programmet innefattade sessioner om internationell utveckling och allmän delaktighet, om utbildning, övning och mätningar, om nödlägen, om kärntekniska anläggningar, om icke-joniserande strålning, om medicinsk användning av strålning, om industriell användning av strålning, om strålningsbiologi, om naturliga strålkällor, om icke kärnkraftanknutet avfall, om NKS (Nordisk Kärnsäkerhetsforskning), om radioekologi och artificiella radionuklider i miljön samt om myndighetsutövning och internationell verksamhet. Förutom inbjudna föredrag och insända bidrag anordnades grundläggande utbildningsföredrag på morgnarna och flera rundabordssamtal. Totalt omfattade mötet nästan 100 bidrag från deltagare representerande minst 10 olika länder. Bredden i ämnesvalen, bidragens vetenskapliga kvalitet samt det intresse mötet väckt återspeglar det goda anseende strålskyddsverksamheter åtnjuter i de nordiska länderna.

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Preface

Nordic cooperation has long traditions. The Nordic Society for Radiation Protection (NSFS) is an associate of the International Radiation Protection Association (IRPA). The members of NSFS are professionals representing all different areas related to radiation protection. Regular meetings are organised every three years. The XIV Regular Meeting will take place in Rättvik, Sweden, 27 – 31 August 2005. As has been the case with the previous two meetings, the organisers are pleased to welcome participants not only from the Nordic countries but also from Estonia, Latvia, and Lithuania.

The theme of the 14th Regular Meeting of the Nordic Society for Radiation Protection is 'Radiological Protection in Transition'. The reason for choosing this theme is obvious. The current recommendations of the International Commission on Radiological Protection (ICRP) were adopted in 1990. New scientific data have since then been published, and the biological and physical assumptions and concepts need updating. It has also become evident that focusing on humans alone is not always sufficient. The system of protection uses a terminology that could be simplified in several aspects, and there have been societal developments in that more openness is expected in developing new recommendations that could be accepted internationally.

ICRP's system of protection has evolved over time as our understanding of underlying mechanisms has increased. ICRP is currently developing a new set of recommendations and initiated a worldwide scientific debate at the 10th IRPA Conference in Hiroshima in the year 2000. The ideas of ICRP have already been published in international journals and promulgated through IRPA, and in 2004, the proposal for new recommendations was discussed at the 11th IRPA Conference in Madrid. The draft recommendations have recently been subject to international consultation. A new draft can be expected in 2006 and a second round of consultation on the recommendations will be necessary. ICRP hopes to adopt the new recommendations late 2006 or early 2007.

It is against this background that the 14th Regular Meeting of the Nordic Society for Radiation Protection convenes in Rättvik, Sweden, to discuss new scientific data of relevance for radiation protection.

The present Proceedings contains some 56 oral presentations and some 39 posters. The Proceedings show the wide range of topics that will be discussed at the meeting and I thank all participants for their contributions to a successful meeting in Rättvik.

LARS-ERIK HOLM
Director General
SSI

Session I: The Lindell lecture

From Pandora's box to Sisyphus – comments on radiation protection yesterday, today and tomorrow

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Abstract: After a short review of the build-up of operational radiation protection and emergency management in Switzerland, lessons learned from these experiences and from regulatory work, international cooperation and public relations are presented. Recent developments and trends are critically discussed.

Introduction

To receive the Bo Lindell Award is a great honour, and I am very grateful to the Board of the NSFS and to its President for this distinction.

A look at today's activities often gives me a kind of "déjà-vu"-feeling. Much knowledge from the "pioneer" period of radiation protection has become lost or forgotten again and some is now re-invented by the younger generation. Part of what we developed from scratch does not seem necessary any more because radiation protection has become so perfect. But if something goes wrong we should remember proven survival techniques.

Pandora's Box: Operational Radiation Protection, Learning from Mistakes

I started my career in radiation protection in 1958 at the Swiss reactor research institute at Würenlingen (EIR, now PSI). As the first full-time radiation protection team in Switzerland we collaborated with the radiation protection and nuclear safety authorities and with commercial nuclear power since the early 60ies. This led to a broad spread of our methods and techniques in radiation protection to the new installations, to a continuous exchange of experiences and to services provided by us. Being primarily an operational division our staff had to deal with many different tasks and problems in parallel which gave them a wide horizon and required flexibility and optimisation of efforts. We aimed at a pragmatic type of radiation protection and practicable solutions for our problems by developing promising ideas and methods from abroad until they worked well in routine situations and could be operated by any technician, not only by the scientist who had invented them. By close contact to producers of instrumentation and protective material they let us test their new products under routine conditions and used our feedback of experiences to improve or develop products according to our specifications. Due to the broad spectrum of activities and installations at Würenlingen we had to solve a variety of protection problems, to measure and analyse a wide range of radionuclides and to deal with radiation fields from background to very high levels. Some of the solutions we developed were meteorological and environmental monitoring instrumentation and methods, ventilated protective suits, air samplers and air monitors, active and passive radon monitoring, personnel dosimetry, incorporation monitoring and analysis from tritium to plutonium. In 1975 we began the first practical application of the effective dose concept for combining external and internal exposures. We improved decontamination methods and optimized conditioning of low and medium radioactive waste in a pilot facility which now has become the national waste management facility.

At a research and development institute some experiment or installation could and did malfunction. Isotope production of short-lived nuclides operating under heavy time pressure was a frequent source of incidents. Contamination incidents of the most varied kinds gave a lot of practical experience in dealing with and preventing contaminations with nuclides from tritium to plutonium. This improved operational protection and monitoring and led to better safety management and to an efficient emergency organisation. Such incidents were not yet considered as criminal offence but as chances to learn and to get more practical experiences in dealing with and preventing incidents, thus improving safety culture. An internal safety council discussed incidents, safety problems and safety aspects of planned installations and operations. In operational protection we switched from the "police" concept (supervising the radiation workers by our technicians) to self-responsibility. We trained the workers in operational radiation protection and the team leaders as protection experts who had to bear the responsibility for the safety of their operations. Our full-time staff concentrated on selective supervision, counselling and dealing with special problems. In 1991 this concept was generalized in the Swiss legislation for all radiation work. Most of the recommendations of the EAN Workshop of September 2004 at Uppsala are already implemented in our legislation of 1991/94.

The Würenlingen school for radiation protection, built up for our own staff, expanded and offered training courses for radiation workers and radiation protection experts for all fields of application. Its curricula and practical training have become standards. The school's medical X-ray training facility is used by the schools for medical x-ray technicians who learn how to optimize the x-ray pictures. Today all medical doctors using X-rays have to get a corresponding practical training.

The development of safety culture both among the users of radiation and in the supervising authorities is fundamental. Mistakes and incidents will always happen, they should not be punished but be openly analyzed and be used to improve training and procedures. Radiation protection should never become too perfect, because then people tend to become complacent and negligent and cause a severe accident. It seems necessary to have a minor incident from time to time, both in order to show to the collaborators that radiation protection rules must be followed strictly, and to demonstrate to the management that the costs of radiation protection and of training are justified!

Damokles' Sword – Emergency Management from Fallout to Tsunami

The early years of radiation protection in Switzerland coincided with the nuclear weapons tests' worldwide fallout and with the build-up of radiation protection in army and civil defence and we became early involved in the protection against nuclear weapons. Military training and experiences were an excellent basis for efficient emergency organisations in nuclear installations and on a national level. The start of the national radiological emergency organisation was caused by the fallout menace, but it was soon expanded to nuclear installations accidents. The FS conference in 1968 had the topic of "Radiological Protection of the Public in a Nuclear Mass Disaster". Switzerland experienced a nuclear reactor core meltdown accident in 1969 at the experimental nuclear power station of Lucens. The reactor was located in a cave, so except a short noble gas leakage the released fission products were contained underground. This was the first occasion where the national emergency organisation and the emergency team of the Würenlingen Institute had to deal with a real accident. This experience and several total defence exercises led to an efficient national emergency organisation comprising all national civilian and military means, prepared to deal with all event types with possible radiological consequences. We

were ready when the Chernobyl accident happened and reacted with all necessary means without delay. After Chernobyl, we became active, together with Jan-Olof Snihs, in the International Nuclear Emergency Exercise (INEX) expert group of NEA which has a pioneer role in improving realistic and modern emergency management for radiological events.

Chernobyl was two decades ago, but knowledge has a half-life of only a few years. Had the lessons from Chernobyl and the INEX exercises really been learned and applied generally to emergency preparedness, the mistakes and delays last December after the Tsunami catastrophe need not have happened. A coordinated and well exercised emergency preparedness is needed for any type of catastrophe. A permanent monitoring and alert team working around the clock and analyzing any potential threat information must be coupled to an acting team which has the competence, means and capacity to immediately deal with the initial phase and to bridge over the time until the national crisis management takes over the lead. There should be only one emergency manager whatever the type of emergency. All national and regional means should be at the disposal of the emergency organization, with an efficient information service and a national communications platform accessible to all collaborating services and authorities. After the start of an emergency, the information needs grow exponentially and must be satisfied although the situation may first be largely unknown and the information input scarce. If a government loses the information battle in the first hours, it loses the confidence of the public and the emergency battle almost as soon as it has begun. Decision-making with insufficient knowledge of the situation must be learned and trained in exercises that even include government members. A comprehensive and flexible legislation must allow and enforce all necessary activities and cooperation in extraordinary situations.

Hercules' Labour: Regulatory Activities – International Cooperation

In Switzerland we include all interested parties in the drafting and consultation processes of new or revised legislation. A Federal Radiation Protection Commission (KSR) is a core of this stakeholder participation. While radiation protection was first regulated under the Atomic Energy Law, the KSR convinced government to create a separate radiation protection law and submitted a first draft. A large group of experts from authorities and from radiation protection in all applications developed the final draft. After three months of public consultation with only slight modification and some parliamentary delays the law came into force in 1991, the revised regulations in 1994. We formulated the law in a comprehensive and flexible way, put the technical details in the regulation and the numerical values into the annexes. ICRP was drafting its new recommendations, and we got enough advance knowledge to fully implement publication 60, by far the first country to do so.

Lawyers and regulators use a language different from that of us scientists and operational experts: We aimed at flexibility for future developments and specific situations, the lawyers requested static and rigid regulations. Many words have different meanings in legal and in daily use. With practical examples we convinced the lawyers of our more appropriate approach. All Swiss legislation must be available in at least in three of the four official languages and for international use in English. A language service tried to put our draft into good and easily understandable German but changed many meanings and interpretations. We improved wordings which could have a double meaning or interpretation, but won the fight to preserve most of our wording. Translating the draft into French and Italian and translation of ICRP or IAEA recommendations from English into our lan-

guages gave similar problems. Many English terms such as 'constraints' cannot be translated adequately into other languages. We try to convince ICRP since decades to be more careful when they create new terms and to provide the correct translations of the major terms into the main languages.

The first generation of staff in our safety authorities came from practical work in nuclear or radiation technology or research and had personal experience with radiation. They had a realistic approach to safety and supervision and exaggerated reactions were rare. Today too many new staff members land directly at an authority desk. Lacking personal experiences, they tend to surrender to the pressure and opinion from politics and media to consider anything nuclear or radiological as basically evil and to react excessively on minor events. The recommendation of last year's EAN Workshop that authority staff should get a practical training in the area they will have to supervise is more than justified.

A very positive aspect of the "consolidation" phase of radiation protection was the build-up of the radiation protection societies and of the international cooperation and exchange of experience, supported by IAEA and NEA/OECD. Switzerland was early interested in international contacts and actively supported the formation of IRPA. Our society, the Fachverband für Strahlenschutz (FS), started as an informal working group of German, Austrian and Swiss radiation protection experts in 1958. We do active work in 14 working groups and at annual congresses of the FS, resulting in over 120 publications. Our journal *StrahlenschutzPraxis* has successfully survived its first ten years. Our congresses use the temporary concentration of experts to discuss problems and produce results for future use. We welcome this change of style also in IRPA congresses. Hopefully IRPA will remove old ballast and become even more active between the International Congresses. We must use the growing possibilities of the internet to spread information and knowledge in our field, but we should watch that the web is not abused.

Sisyphus' Frustration – Public Relations, Bureaucracy and Politics

I detect a growing frustration among my active colleagues, not because radiation protection would have become less interesting but because of a deterioration of the working environment. Bureaucratic requirements are growing and take more time without improving protection. Political correctness and media pressure dominate discussions and decisions. Public relations remain an unsolved problem, and despite better and more efficient technical means we still have only limited success. All I discussed in my paper on information of the public at the 1973 IRPA congress is still valid. A military style analysis of the situation before confronting the public in a PR activity or making a decision in routine or emergency situations is indispensable.

The authorities lack real problems because protection has become so perfect. To justify their existence and size some tend to create new "problems" using legal and bureaucratic tricks. Radiation protection has no own political and medial basis and support to fight the increasing pressure from politics and media. The former nuclear research institutes have disappeared or changed their programs. For radiation protection this has removed or reduced the possibilities for practical experiences and developments. The formalistic research programs of the EU are no real replacements. One of the positive aspects is the improving international cooperation, especially within IRPA, and the growing opening of radiation protection towards safety at the working place, accident prevention, general emergency management and environmental protection, where it can play a leading role with proven concepts.

Session II: International developments and stakeholder involvement

The new ICRP recommendations on radiological protection

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Abstract: The present recommendations of the International Commission on Radiological Protection (ICRP) were adopted in 1990. Since then, ten publications have provided additional guidance for the control of exposures from radiation sources and they specify some 30 different numerical values for restrictions on individual dose for differing circumstances. ICRP is currently developing a new set of recommendations that will consolidate all the advice included in and developed since Publication 60. When ICRP adopts the new recommendations in 2006 or 2007, about 16 years would have passed since the 1990 recommendations were adopted.

Introduction

The primary aim of ICRP's recommendations is to provide an appropriate standard of protection for human beings, and where necessary for other species. ICRP's recommendations were revised substantially in Publication 60 (ICRP, 1991). Since then there have been many additional numerical restrictions on dose, based on different ideas, and spanning several orders of magnitude (ICRP 1993a-c, 1994, 1995, 1997, 1998a-b, 1999). A framework for environmental protection has also been published (ICRP 2003a). New scientific data have been published since Publication 60, and the biological and physical assumptions and concepts need some updating. The terminology used can also be simplified in several aspects, and more openness is expected in developing new recommendations that could be accepted internationally. All this combined make the time ripe for a new set of recommendations, while recognising the need for stability in international and national regulations, many of which have only relatively recently implemented the 1990 recommendations.

ICRP has prepared a series of building blocks on which to base the new recommendations. Five such documents were put on ICRP's website for consultation in April 2005. They deal with biological and epidemiological information on health risks, the basis for dosimetric quantities, assessing doses to the representative individual, the optimisation of protection, and the concept and use of Reference Animals and Plants.

The international consultation on the draft recommendations was the culmination of several years of work and resulted in nearly 200 responses with some 600 pages of written text. Many comments necessitate some clarification of policy points, but most of the comments deal with issues that will be explained in the building-block documents, which were not available at the time of consultation. As a result of the consultation, two other documents are considered necessary. One will deal with the protection of the patient in medical procedures, and the other will be an updated version of Annex C in Publication 60 (ICRP, 1991). Annex C demonstrated that ICRP does not have a simple risk-based system, but rather that there is a complex multi-attribute assessment of the implications of exposure and a valid comparison with the natural background radiation levels.

Qantities and Radiation Weighting Factors

Equivalent dose will change name to radiation-weighted dose in order to avoid confusion with dose equivalent and reduce problems in translations from English to other languages. Effective dose remains the principal quantity for radiological protection.

The radiation-weighting factors (w_R) have been revised (ICRP, 2003b). For photons and beta particles, a w_R of unity is retained for all low-LET radiations, and a w_R of 20 is retained for alpha particles. ICRP believes that a w_R of 5 to all protons of energy > 2 MeV is a significant overestimate of the biological effectiveness of these protons. Therefore, a w_R of 2 is proposed for incident protons of practical importance (> 10 MeV). For neutrons, ICRP recommends a continuous function, and the w_R for neutrons should be decreased for energies below 1 MeV to take account of the dose contribution from gamma rays that are induced in the body by neutrons.

Radiation Effects and Tissue Weighting Factors

Current understanding of mechanisms and quantitative data on dose and time-dose relationships support a linear dose response at low doses for total cancer risk. A better understanding of the mechanisms for radiation-related adaptive response, genomic instability, and bystander effects is needed before they can be evaluated as factors to be included in the estimation of risk after low doses. While existence of a low-dose threshold does not seem unlikely for radiation-related cancers of certain tissues, the evidence as a whole does not favour the existence of a universal threshold, and there seems to be no particular reason to factor the possibility of a threshold into risk calculations. A dose and dose-rate effectiveness factor of 2 is retained. The linear, no-threshold hypothesis remains a prudent basis for radiation protection at low doses and low dose rates.

New radiation detriment values and tissue weighting factors have been proposed; the most significant changes from ICRP 60 relate to breast, gonads and treatment of remainder tissues (Table 1).

Table 1. Proposed tissue-weighting factors.

Tissue	w_T	$\sum w_T$
Bone-marrow, colon, lung, stomach, remainder tissues[1] (nominal w_T applied to the average dose to 15 tissues)	0.12	0.60
Breast, gonads	0.08	0.16
Bladder, oesophagus, liver, thyroid	0.05	0.20
Bone surface, brain, salivary glands, skin	0.01	0.04

[1] **Remainder tissues (15 in total:** Adipose tissue, adrenals, connective tissue, extrathoracic (ET) region (as defined in ICRP Publication 66), gall bladder, heart wall, kidneys, lymphatic nodes, muscle, pancreas, prostate, small intestine wall, spleen, thymus, uterus/cervix.

[2] w_T for gonads is applied to the mass-weighted mean of the doses to testes and ovaries (i.e. the average dose in gonadal tissue).

Detriment-adjusted nominal probability coefficients for cancer are $5.9\ 10^{-2}$ Sv^{-1} for the whole population and $4.6\ 10^{-2}$ Sv^{-1} for adult workers, as compared to the values in ICRP

60 of 6.0-10^{-2} Sv^{-1} and 4.8 10^{-2} Sv^{-1}, respectively (Table 2). The detriment-adjusted probability coefficients for hereditary disease up to the second generation are 0.2 10^{-2} Sv^{-1} for the whole population and 0.1 10^{-2} Sv^{-1} for adult workers; the respective ICRP60 values are 1.3 10^{-2} Sv^{-1} and 0.8 10^{-2} Sv^{-1} but these relate to risks at equilibrium. Risks of non-cancer disease at low doses remain uncertain and no specific judgement is possible.

Table 2. Nominal probability coefficients for cancer and hereditary effects (10^{-2} Sv^{-1}).

Exposed population	Detriment adjusted cancer risk	Detriment adjusted heritable effects	Detriment	Detriment in Publication 60
Whole population	5.9	0.2	6.1	7.3
Adult workers	4.6	0.1	4.7	5.6

Principles of the System of Radiological Protection

The draft recommendations made attempts to clarify the role of governments and protection professionals in justifying the use of radiation sources. There was concern expressed that the principles of protection were not clearly described, and ICRP will clarify that the basic principles of radiological protection are retained in the new recommendations. In Publication 60 the three principles were expressed separately for practices and intervention. ICRP now proposes one set of principles. The principles of justification and constrained optimisation are source-related and apply to normal, emergency and existing controllable situations. The principle of dose-limits is individual-related and applies to normal situations but not to emergency situations to existing controllable exposures. For occupational exposure, they apply to those situations, which are determined by the regulator.

The concept of the *constraint* was introduced in Publication 60 (ICRP, 1991) as part of the principle of optimisation of protection: '*This procedure should be constrained by restrictions on the doses to individuals (dose constraints), or the risks to individuals in the case of potential exposures (risk constraints), so as to limit the inequity likely to result from the inherent economic and social judgements*'. The concept was not clearly explained by ICRP in its subsequent publications, and ICRP now aims to clarify the meaning and use of constraints.

In practice, for any source in any situation, the regulator will have some level of individual dose beyond which the regulator will not sanction actions by the operator. This level is to be seen as the constraint for the situation under consideration and is the upper bound to individual dose in optimising protection in that situation. The result of the optimisation process is to find the authorized level for planned operations, the intervention levels for protective actions in emergencies or the clean-up level for contamination. These levels will generally be lower than the respective constraint. ICRP will recommend a basic set of three constraints that covers all situations to be addressed (1, 20 and 100 mSv). These include the occupational and public doses in normal operation, accidents and from contaminated land. Comforters and carers of patients undergoing therapy with radioactive sources are included.

Discussion

The probabilistic nature of stochastic effects makes it impossible to make a clear distinction between 'safe' and 'dangerous', and some finite risk must be accepted at any level of protection. This has led to the System of Protection with its principles of justification, constrained optimisation and dose limits. These principles will continue to be the cornerstones of ICRP's recommendations.

The new recommendations should be seen as consolidating the recommendations in Publication 60 and those published subsequently, to give a single unified set that can be simply and coherently expressed. The opportunity is also being taken to introduce a clear and parallel approach for the radiological protection of the environment.

The next draft of the recommendations will be completed after the finalization of the building blocks and should be ready for Main Commission consideration in 2006. A second round of consultation on the recommendations will be necessary, after which ICRP will need to complete them. The most likely consequence of this will be that the publication of the new recommendations will not be adopted until late 2006 or early 2007.

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Council Directive 2003/122/Euratom on the control of high-activity sealed radioactive sources and orphan sources (HASS-directive) and its implementation in Sweden

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Abstract: The Council Directive 2003/122/EURATOM on the control of high-activity sealed radioactive sources and orphan sources (HASS-directive) was adopted on the 22 December 2003 with the purpose to prevent exposure of workers and the public to ionising radiation arising from inadequate control of high activity sealed radioactive sources or orphan sources. The Directive also has the aim to harmonise controls in place in the Member States by setting out specific requirements ensuring that each such source is kept under control. The Member States shall implement the Directive before the 31st of December 2005.

Introduction

Radioactive sources are used throughout the world for a wide variety of purposes in industries, nuclear facilities and for medical and research use. The risk associated with the use of such sources varies widely depending on their activity, radionuclide content and construction, but the risks associated with the planned use are normally well known. The use of radioactive sources in the European Union is subject to legislation and regulations based on the Council Directive 96/29/EURATOM laying down basic safety standards for the protection of the health of workers and the general public against the dangers arising from ionizing radiation (BSS-directive) [1]. For example in Sweden through the Swedish Radiation Protection Act [2] and Ordinance [3] together with regulations issued by the Swedish Radiation Protection Authority.

A number of events involving sealed radioactive sources out of control during the last decades have pointed on the problem with sources not under control. Such "orphan sources" may lead to serious radiation injuries or even deaths as for example in Brazil, Georgia and Estonia. Findings of sources in scrap yards and metal production facilities are recurrent events in most countries and may be dangerous for the persons handling the goods, cause releases to the environment, contamination of the facility and of the metal produced with high costs for monitoring, cleaning up and decontamination.

Based on these facts the European Commission stated that it would be appropriate to strengthen the control on those radioactive sources posing the greatest risk and emphasize the responsibilities of holders of the sources. Prevention of radiological accidents and injuries requires the location of each high activity source to be known, recorded and verified from the time the source is manufactured or imported into the Community until the source is placed in a recognized facility for its long term storage or disposal or is exported from the Community – "from cradle to grave". Also changes in the location or use must be recorded and notified to the Authorities. It is also considered important that similar ways of control is used in all member states of the Community as sources may be moved between the countries.

In 1999 the European Council therefore took the decision that there was a need for the European Union to address the problem and the work to write new legislation started. A proposal together with background material was published [4] and the directive was adopted 22 December 2003 [5].

The same problem has also been investigated by other international organizations, especially the International Atomic Energy Agency (IAEA), where also antagonistic aspects were included. That work resulted in development of "Code of Conduct on the Safety and Security of Radioactive Sources"[6] together with "Guidance on the Import and Export of Radioactive Sources"[7]. Sweden has in writing taken the position that the provisions in the Code of Conduct should be followed.

Main provisions of the HASS-directive

For the purpose of the Directive a "High Activity Sealed Source" (HASS) is defined as a sealed source containing a radionuclide whose activity at the time of manufacture or, if that is not known, at the time of the first placing on the market is equal or exceeds a listed activity level. That level corresponds to one hundredths of the A_1–value in the IAEA Regulations for the safe transport of radioactive material [8]. The dose-rate at 1 m from an unshielded HASS is approximately 1 mSv/h.

Article 3 of the Directive requires that each holder of a HASS shall have a prior authorization for any practice involving the source, including taking possession of the source. Before issuing an authorization, authorities must ensure that adequate arrangements for the safe management of sources have been taken, including when they become disused sources. Such arrangements may provide for the transfer of the source to the manufacturer or supplier or to a recognised installation. A "recognised installation" means a facility, which is authorized for the long-term storage or disposal of sources or for the interim storage of sources. When placed in such a facility the source should never come into use again. Before issuing an authorization the authority also has to be ensured that adequate provisions of financial security or other equivalent means have been taken for the safe management of the source, including the case when the holder becomes insolvent or goes out of business.

Authorization should cover responsibilities, minimum staff competencies, requirements for emergency procedures and communication links, work procedures, maintenance and management of disused sources including the transfer to the manufacturer, supplier, another holder or a recognised facility.

Article 4 requires that the authority be informed about each individual transfer of sources.

According to article 5 the holder is required to keep records of all sources under his responsibility, their location and their transfer. The records shall include specific information set out in a standard record sheet, which shall be sent to the competent authority at least once every year or when a situation is changed. The competent authority is required to keep records of all authorized holders and of the sources they keep.

Article 6 requires the holder to regularly arrange for suitable tests, such as leak tests, to maintain and verify the integrity of each source, to establish procedures to prevent unauthorized access to the source and preventing loss, theft or damage by fire.

Manufacturers of sources are required to identify each source with a unique number, which shall be engraved or stamped on the source and on the source container. The manu-

facturer shall also ensure that the source, if possible, and the source container are marked with the warning sign. The manufacturer shall also provide a photograph of each manufactured source type, the source container, transport packaging, device and equipment. If the manufacturer is located outside the Community, the supplier must ensure that this information is available.

In addition to the training and information already required by the BSS-directive the holder shall ensure that the training also includes specific requirements for the safe management of HASS.

Concerning orphan sources each Member state shall ensure that

- National authorities are prepared to take measures to recover orphan sources and to deal with radiological emergencies due orphan sources,
- technical advice and assistance are made available to persons that are not normally involved in operations subject to radiation protection requirements, but who suspect the presence of an orphan source,
- systems aimed at detecting orphan sources in places as large metal scrap yards and major metal scrap recycling installations or nodal transit points, such as customs posts, are established,
- campaigns are organized to recover orphan sources left behind from past activities, and
- a system of financial security to cover intervention cost relating to the recovery of orphan sources is established.

Member states shall bring into force the laws, regulations and administrative provisions necessary to comply with the Directive before 31 December 2005. For sources placed on the market before that date member states may decide that parts of the directive does not apply until 31 December 2007.

Implementation in Sweden

The transposition of the HASS-directive into Swedish legislation will be performed by different means and the responsibility will be shared between the Ministry of Sustainable Development and the Swedish Radiation Protection Authority (SSI).

Amendment of the Radiation Protection Act and Ordinance
The Radiation Protection Act and Ordinance [2,3] have to be amended to include requirements for financial security for the safe management of sources when they become disused sources. A proposal to amend the Radiation Protection Act and Ordinance is under preparation by the Ministry of Sustainable Development, but a decision by the Parliament is not expected until late this year.

Ordinance with Instruction for the Radiation Protection Authority
Designation of a competent authority is under preparation and a decision by the Government to designate SSI as the Competent authority is expected later this year.

Regulations and administrative provisions
SSI has the mandate to issue regulations and administrative provisions based on the Radiation Protection Act. New regulations and amendments of current regulations are under development. These will cover:

- Detailed requirements on authorisation (Art.3)
- Recordkeeping and reporting (Art.4-5)

- Detailed requirements for holders (tests, security measures, return of disused sources, notifications to the competent authority) (Art.6)
- Identification and marking (Art.7)
- Training of personnel (Art. 8)

A proposal for new regulations was sent out for consultations in June 2005 and comments are required before mid August. SSI will continue with the legislative work immediately after comments have been received and the new regulation will enter into force 1 January 2006.

Other issues
SSI is working with the extension of its record system and database to include not only information on licensed holders and approved equipment but also details on each source covered by the Directive. The recording system will be tested during autumn and taken into use from 1 January 2006.

Guidance material in the form of pamphlets and web based information to holders, personal at customs stations and metal recycling plants are under development.

Preparation of emergency preparedness plans to include advice, assistance and response in case of orphan sources is also under development.

Conclusions

After implementation of the HASS-directive in all EC member states together with provisions included in the IAEA Code of Conduct for the Safety and Security of Radioactive Sources [6,7] a better control over high activity radioactive sources will be obtained with a lower risk that sources come out of control and become orphaned.

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EAN - the European ALARA Network

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EAN - the European ALARA Network

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Fortum AB and Norwegian Radiation Protection Authority

The European ALARA Network, EAN, was created by the European Commission in 1996 to facilitate and promote further specific Euro-pean research on the optimization of radiation protection regarding all types of occupational exposure and the dissemination of good ALARA pratices within all sectors of European industry and research. Today EAN has become a mature platform for exchange of prac-tical experience and competence in radiation protection. EAN is a self-governing entity financed by contributions from the participating countries. Its activities are planned by a steering group representing 18 European countries of which four is Nordic. Centre d'etudes sur l'evaluation de la Protection dans le domaine Nucléaire (CEPN) is responsible for the coordination of the network activities.

At first most participants in the network activities were radiation protection experts from regulatory bodies, nuclear research centres and major utilities. Gradually other participants have joined the network activities e.g. representatives from European societies for Non Destructive Testing, Medical Radiographers Associations, Medical Physicians Associations, Medical Physicists Associations, manufac-tures of devices using ionizing radiation, manufactures of radiation monitoring instruments and trainers in radiological protection. Also representatives of international organizations such as ILO and IAEA have participated.

Sub-networks have been formed to deal with specific topics or sectors. One sub-network is focusing on nuclear research reactors another is focusing on developing and maintaining codes of pratice for NDT operators and NDT clients.

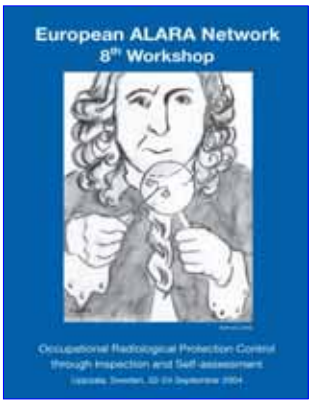
Main objectives

- **to maintain, enhance and develop competence in radiation protection with the special emphasis on the implementation of the ALARA principle for occupational, public and patients exposure**
- **to contribute to the harmonisation of radiation protection policies and pratices both at regulatory levels and operational levels within European countries**
- **to contribute to the integration and effective co-operation of expertise in radiation protection that is available in the countries**
- **to cover radiation protection themes relevant to all sectors**

Main activities

- **ALARA workshops**
- **The European ALARA Newsletter**
- **The EAN web site**

Visit EAN at
http://ean.cepn.asso.fr



Workshops

EAN aim at organising a workshop within the ALARA domain once a year. The workshops (50-80 persons) are forums for the exchange of experience, discussions and networking. At the workshops presentations of invited papers alternate with discussions in small working groups (5-10 persons) where all the participants are expected to contribute. Main conclusions are drawn from the presentations and the discussions in the working groups. At the end of a workshop the working groups propose to all the participants a set of recom-mendations that they would like to issue and to whom the recommendations should be issued to. After validation by the EAN steering group the recommendations are relayed. About 10 recommendations are issued from each workshop. Most of the recommendations has been sent to national regulatory bodies and to the European Commission. The title of the next workshop is *Exposures from NORM and Radon in Workplaces* and the workshop will take place in Augsburg, Germany 18-21 October 2005.

Methods and techniques to arrange stakeholders' decision making workshops in radiological emergencies

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Abstract: International organisations have for many years recommended that key players, e.g., authorities, expert organisations, industry, producers of foodstuffs and even the public should be involved in planning of protective actions in a case of nuclear accident. Often the same stakeholder groups are also involved in real decision making on off-site countermeasures to protect the affected population. Arranging of workshops, where representatives from various fields of the society aim to identify feasible countermeasures in a radiological emergency situation, needs special working methods and decision analysis techniques in order to find out the most feasible countermeasure strategies and to make decision making process transparent and auditable. This paper provides experiences gained in several European countries on how to facilitate this kind of workshops or negotiations and how modern decision analysis techniques can be applied in decision making in nuclear emergencies.

Decision making in off-site nuclear emergencies

Openness, transparency and participation of the key players are important factors for balanced decision-making in public issues. The research in key players' involvement in environmental decisions has lead to the conclusion that if relevant parties are not engaged in the decision-making process the policy will fail. Also international organisations in radiation protection have recognised the importance of prompt, open and transparent decision-making based on scientific facts and social judgements. They have stressed that the basis of the decision must be perceived by the public and all relevant factors concerning the decision should be considered in a rational manner.

The planning and implementation of protective actions in a case of nuclear or radiological emergency is co-operation of several authorities and expert organisations. Composition of the groups planning decisions on countermeasures depends on the type of an emergency and also on the phase of the situation. For example in a case of a severe nuclear power plant accident, there is always a certain time before any radioactive releases to the environment take place. In this threat phase, decisions are normally made by the operator and the local rescue officers, and the decisions are based on 'best estimates' about the development of the plant condition. In later phases of the accident also other organisations will be involved in decision making and planning of countermeasures or protective actions. Also the grounds of decisions will change when more information about the accident and the radiological situation in the environment is available.

To be able to make right decisions at right time we have to know the whole emergency management process. Another requirement for proper emergency management is that we have operational systems to trigger the emergency management and to collect information about the emergency situation (emergency plans, monitoring systems, decision support systems). Furthermore we should have methods to rank different countermeasure options in a way that the decisions are transparent and traceable (evaluation tools, decision

analysis). Finally, because various stakeholders are involved in decision making in different phases of the situation, we need to have methods to get impact from all relevant stakeholders in a reasonable time (facilitated workshops).

Decision support systems and countermeasure evaluation tools

At the moment RODOS is the most comprehensive decision support system (DSS) developed under the umbrella of the EURATOM Research Framework Programmes. Another system having similar features is the ARGOS system developed mainly by the Danish Emergency Management Agency (DEMA). All DSSs should provide so called 'hard factors' to decision makers, i.e. radiological data, meteorological data, dose assessments, dispersion prognosis, demographic data, etc. However, in real life also softer values will be incorporated in decision making of countermeasures, especially in later phases of a radiological emergency. These subjective factors can relate to political acceptability, public reassurance and trust, equity, socio-psychological consequences, etc. depending on the stakeholders being involved in decision making.

Decision support systems, in order to provide real support to decision makers, should be able to handle also these 'soft values' together with the more measurable 'hard factors'. Some kind of decision analysis tools are needed to be able to make trade-offs in relation to the emphasis given to e.g. public perception and political acceptability against more objective criteria, such as radiation health effects and costs. Modern decision analysis tools are now integrated with both RODOS and ARGOS decision support systems. This integration was done in the EVATECH project (*Information Requirements and Countermeasure Evaluation Techniques in Nuclear Emergency Management*) in the fifth EURATOM Framework Programme (FP5). A generic Java-based multi-attribute evaluation software WEB-HIPRE (HIerarchical PREference analysis in the World Wide Web) was integrated with RODOS system and a standard windows-based multi-attribute evaluation tool VISA (Visual Interactive Sensitivity Analysis) was more loosely integrated with ARGOS system. In the same project, WEB-HIPRE was also tested in nine facilitated workshops in six European countries. In the workshops real stakeholders tried to find most feasible countermeasure strategies to clean-up contaminated urban environment.

Use of Web-Hipre and RODOS in facilitated workshop

The multi-attribute decision analysis tool WEB-HIPRE can be used either by a single user or in a group consisting of representatives of relevant stakeholders of the problem to be decided. WEB-HIPRE was used in several facilitated workshops focusing on clean-up actions in inhabited areas after a nuclear accident. The participants of the workshops represented real decision making organisations such like ministries, provincial governments, affected cities, waste management, consumer services, police, rescue services, nuclear power companies, radiation and nuclear safety authorities, health authorities, food control, etc. Aims of the workshops were to develop methods for stakeholder involvement in decision making, to verify the factors driving decision making, and to explore the information needs of the participating stakeholders. Intention was also to explore how uncertainties could be incorporated in decision making.

The first task in this kind of workshops is to define the factors (attributes or values) which are driving the decision making. The factors are related to the environmental radiation

situation, health effects of radiation, mental health effects, psycho-social consequences, socio-political issues, economic factors, etc. The next step is to consider what countermeasures are available for protection of the public and how feasible these measures are. Different potential countermeasure strategies can be compiled for more detailed evaluation. In ranking of the strategies the participants have to make difficult trade-offs between 'objective' and 'subjective' criteria. However, the adopted evaluation tool WEB-HIPRE enables this kind of ranking in a relative simple way. The aim is to find the most feasible countermeasures which can be also explained to the general public in a transparent and traceable way.

The facilitator is a key person in this kind of discussions. His/her role is important and demanding because a facilitated workshop is an interactive event. The facilitator is not a chairperson or a group leader. He/she should not share the problem and should concentrate on the process, not on the content which is left to the participants. The facilitator is a person rising issues neutrally and asking simple questions to ensure that all have understood the point.

Figure 1 shows an example of the value tree created with WEB-HIPRE in one of the facilitated workshops performed in EVATECH. The rightmost column contains protective actions selected by the participants. The alternatives are the selected strategies containing one or more of the potential protective actions. The next column (Criteria 2) shows the lower level attributes selected by the participants. These lower level attributes are further grouped into the higher level attributes (Criteria 1). The leftmost column indicates the overall coal of the decision making panel, in this case it is called as 'normal living conditions'. The panel has to identify the strategy which leads as close to the overall objective as possible, taking into account the values of the participants.

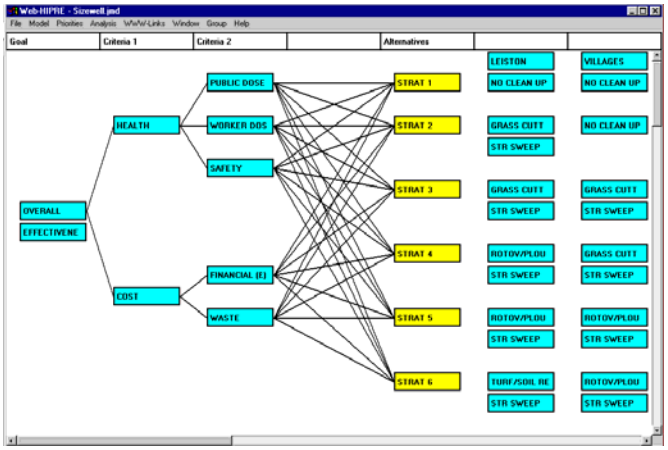

Figure 1: Example of the value tree produced at one of the workshops of EVATECH project.

The values or criteria (also called as attributes) discussed in the nine stakeholder workshops were surprisingly similar. The main groups under which almost all the discussed criteria can be classified were health related issues (public/workers' radiation doses, workers' physical safety, etc.), social/political aspects (political acceptability, public reassurance and confidence, socio-psychological effects, equity, environmental protection, etc.), and technical feasibility (costs, available resources, waste management, etc.). The mutual weights of the criteria understandably varied in these workshops depending on the level of contamination being used in the scenario, the total area to be

cleaned up, resources available and the number of people affected. If the contamination and the consequent radiation doses were not high, social and political aspects got a greater weight than health related issues: on the other hand, if the contaminated area was wide, technical feasibility and costs had a greater importance. As an example, Figure 2 shows how the different criteria contributed to the scoring of the strategies discussed at one of the EVATECH workshops.

Uncertainties associated with the different attributes in decision making in management of off-site nuclear emergencies can be treated with the sensitivity analysis tool of WEB-HIPRE. This brings an important added value because uncertainties in the model predictions of the radiation exposure can be several orders of magnitude.

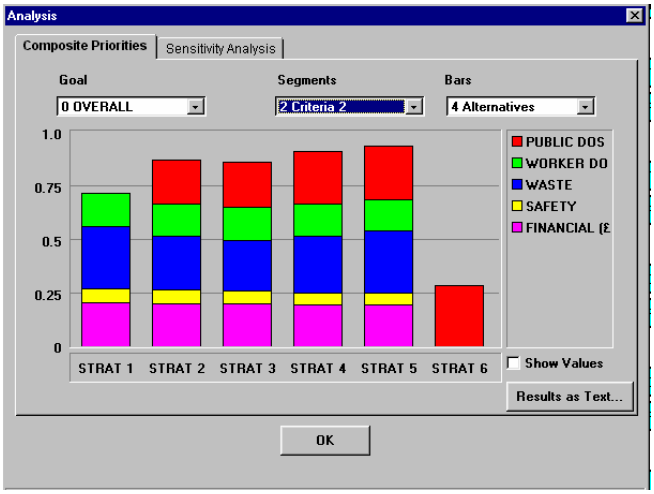

Figure 2: Composite priorities of the strategies discussed at one of the EVATECH workshops, showing the proportion of scores on different attributes

Discussion

The experience gained strongly supports the format of a facilitated workshop for tackling a decision problem that concerns many different key players. The participants considered the workshop and the decision analysis very useful in planning of actions in advance. They also expected a similar approach to be applicable in a real situation, although its suitability was not rated as highly as for planning. The suitability of the approach in the early phase of an accident was rated the lowest. It is concluded that a facilitated workshop is a valuable instrument for emergency management and in exercises in order to revise emergency plans or identify issues that need to be resolved.

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Stakeholders engaged in restoration of the food supply chain after accidental radionuclide contamination

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Abstract: Five European countries participated in the network project FARMING (www.ec-farming.net) in the 5th Framework Programme of the EC. Stakeholders representing the food supply chain from 'field to fork' were invited to participate in evaluation of rural countermeasures and related waste disposal options. The functioning of the Finnish group, its values, goals and documented results suggest appreciation of the group's potential for important contributions to emergency preparedness planning. The main achievements were the identification and formation of a motivated, well communicating group, and, resulting from group processes, an update of relevant multidisciplinary knowledge base and views concerning practicable countermeasures.. Clarifying the status of the Finnish group would ensure the continuance of effective stakeholder engagement and sufficient level of competence for such a group's active role in planning and implementation of practicable intervention strategies.

Introduction

Stakeholders (sagägare, delägare, intressent; asianosainen, osapuoli) are those having a stake in an organisation or activity. The synonym is not an interest group, although such a group can be included in stakeholders. Various organisations have not always identified or explicitly considered all the stakeholders they have impact on. Concerning the food supply chain, stakeholders are actors that are responsible for the functioning of the food supply and for control of food safety. They may also be people influenced by decisions on food production and supply.

Particularly since the 1990s stakeholders have been considered in the field of radiation protection, environment, and emergency preparedness of various kinds, etc. The OECD organised three Villigen workshops for presentations of case studies and discussions about the role and practices of stakeholder participation in radiation protection (OECD, 2004). In United States emergency preparedness of nuclear power production has initiated stakeholder participation (Nevin Dressler, 2004) and partnership programmes for food security (Graham 2004). Increasing number of advisory boards and environmental forums are encouraged to activate stakeholder participation in decision-making. Stakeholder engagement is being examined, and its value for decision-making with societal dimension analysed. Guidance on who the stakeholders are and why and how they should be involved is provided by international organisations (MSH, UNICEF, 1998). To achieve more than through simple consultation of external experts, it is essential to find the best methods for working and coordination.

Many issues of food safety often have a strong social dimension, but particularly in connection of radionuclide contamination of foodstuffs there is a great demand for consideration of stakeholders. CEC's White paper on safety of food gives emphasis on stakeholder involvement and its transparency (EFSA 2003).

National stakeholders representing the food supply chain 'from field to fork' were identified and invited to participate in the thematic network FARMING under the EC's 5th Framework Programme (www.ec-farming.net). The objective was evaluation of agricultural countermeasures and related waste disposal options (Nisbet et al. 2005) using working material of the project STRATEGY (www.strategy-ec.org.uk).

The aim of this paper is to characterize Finnish stakeholder engagement under the FARMING project from the perspective of the national coordinator.

Results and discussion

The Finnish stakeholder organisations of the food supply chain were identified and their representatives nominated in early 2001. In addition to the coordinator, Radiation and Nuclear Safety Agency, eighteen organisations became involved. The main activity during the first three years was the annual seminar where topical discussions followed expert presentations or other communication necessary for the group. The group gradually prepared for the second year task, evaluation of practicability of 30 countermeasures and a dozen waste disposal options. The seminar reports 2001-2003 are found on the web pages of the project (www.ec-farming.net).

For evaluation of countermeasures the Finnish group agreed on the general criteria for practicability given for the project: radiological impact, technical feasibility, capacity (workers, machinery, time available etc.), costs (working time, use of machines, materials), environmental impact, and acceptability The group also included secondary costs (wastes etc.) and social and economic impact in the evaluation. Northern production conditions and practices were of special concern to the group. The evaluation process resulted in a considerable knowledge base for the group, and certainly improved the competence of the members for investigation and planning of countermeasure strategies.

Coordination of stakeholder activities through the steering group of the project was important for exchange of information between the participants from the five countries. The WISDOM workshop in 2003 provided a forum for stakeholder discussions and presentations, and for voluntary contributions to proceedings (Enqvist; Orre; Rantavaara et al. (b); Root et al.; 2005). The group also initiated a communication plan for households and a research project on costs of clean feeding in dairy farming (Rantavaara et al. 2005a). In addition, group members were active in writing in customer magazines of their own organisations and planning the use of their own communication networks for dissemination of information on issues related to intervention. Training of farmers in planning and implementation of countermeasures was suggested by the agricultural experts in the group, and support to households was also suggested.

Finnish stakeholder participation in the project FARMING was a positive experience. Robust analysis of the group's interests and impact, and coordination strategies used show the importance of basic principles of working as a group (Table 1). At an early stage of the project the terms of reference were formulated together. The terms summarised values and content of interactive working, consideration of communication, high quality information and the local stakeholders. The terms of reference were found relevant and gave a meaningful support for coordination during the four year project.

Advantages of a small country are, for example, easy connections between expert and administrative organisations and industrial life. Having had no serious food crisis in the

recent history was also an important factor that was reflected as confidence in authorities and acceptability of reasonable strategies to maintain the food supply.

There is a risk that motivation and competence of the stakeholder group is loosened after closing of the initial project that responded to the actual need for bringing stakeholders together and combining resources through networking. To avoid loosening of the existing good opportunities, the status of the group as an advisory team in the national emergency preparedness organisation should be defined. An actively contributing group can effectively support authorities and other decision-makers to face the challenges of an unexpected food crisis.

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Table 1. Analysis of the Finnish stakeholder group in the thematic network FARMING

Stakeholders from governmental organisations	Stakeholder interests	Group impact	Coordination strategies
Ministry of Agriculture and Forestry, *Department of Agriculture, Department of Food and Health*	Chairmanship & ⇩	Clarified responsibilities	Briefing chairpersons
Ministry of Social Affairs and Health, *Department for Promotion of Welfare and Health*	Communication	Communication	Collaboration Communication
Ministry of Trade and Industry, *Trade Department*	Contributing	Emphasis on responsibilities of own organisation	Documentation of stakeholder contributions
National Food Agency			
Joint Municipal Authority for Health Care in Loviisa District	Improving competence in emergency preparedness	Improved competence	Emphasis on objectivity, openness, quality of work
Finnish Food and Drink Industries' Federation (FFDIF)			
Plant Production Inspection Centre, *Agricultural Chemistry Department*		Initiatives for training, communication and research	Information from the steering group
Finnish Environment Institute, *Impacts Research; Agriculture, Forestry and Catchment Processes Unit*	Learning from other stakeholders		Presentations of experts
National Emergency Supply Agency		Respect for terms of reference	Providing literature for background
Non-governmental organisations			
Association of Rural Advisory Centres (ARAC, ProAgria)	Networking		
Finnish Association of Agricultural Journalists		Shared values and goals in improving quality of preparedness	Public information of the project
Foodstuffs Industry Pool	Preparedness of own organisation		
Finnish Food and Drink Industries' Federation (FFDIF)			Requests for contributions & clarifications
The Martha Organization (a Home Economics Extension Organization)		Using own networks for dissemination of information	
Central Union of Agricultural Producers and Forest Owners MTK	Profiling		Respect for time of stakeholders
Finnish Food Marketing Association			Sharing knowledge
The Finnish Consumers' Association	Sharing expertise	Voluntary contributions	
Finnish NGO's Agriculture Task Force Birdlife Finland			Trust in group
STUK, Ecology and Foodchains	**Coordination**	⇨	

Session III: Education, training and measurements

Surveillance of radioactive discharges from Institute for Energy Technology (IFE)

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Abstract: The Environmental Monitoring Section of the Health and Safety Department at IFE performs surveillance of discharges of radionuclides to air and water from IFE´s nuclear activities at Kjeller. The main tasks of the surveillance programme are to monitor the discharges of radioactivity and control that these do not exceed the limits given by the Norwegian Radiation Protection Authority (NRPA), and to monitor the levels of radioactivity in the local environment (recipient). This presentation contains details about the monitoring programme.

Introduction

IFE operates a 2 MW (thermal output) research reactor (JEEP II) at Kjeller. The low level liquid radioactive waste originating from this and other nuclear activities at IFE is discharged to the Nitelva River. There are also discharges to air from these activities. Discharge of radioactive substances from IFE´s activities at Kjeller is regulated in a permit given by NRPA. The permit for discharge demands an annual recipient control to determine concentrations of discharged radionuclides in the environment. The Environmental Monitoring Section is responsible for the continuous monitoring and control of the discharges and for the annual report of the results to NRPA (Nystuen and Raaum, 2004). In addition, the section is responsible for monitoring possible release of radionuclides from the Norwegian repository for low level radioactive waste (KLDRA-Himdalen).

IFE has designed a monitoring programme that has been approved by NRPA. One part of the programme is the discharge control. The other part is the recipient control which involves an extensive sampling and analysis of areas whitin and around IFE. All procedures used in the monitoring programme are specified in the Section´s Quality Assurance Manual (IFE 2004).

Discharge and recipient control at IFE, Kjeller and the surrounding areas

Surveillance of discharges to air and recipient control

Samples of both outdoor air and discharged air from active areas are collected on coal filters. In addition rainwater, grass, milk and agricultural samples are collected both inside and outside of IFE (Table 1).

Table 1. Surveillance of discharges to air

Sample	Number of sampling sites	Sampling sites
Outdoor air	2	Within IFE´s area
Discharges to air	3	Within IFE´s area
Rainwater	5	Within IFE´s area
Grass	5 6	Within IFE´s area Outside IFE´s area
Milk	2	Local farms
Agricultural samples	2	Local farms

Outdoor air:
Outdoor air is under continuous surveillance at two sampling sites within IFE. The samples are analysed for gamma emitters.

Rainwater:
Rainwater samples are collected once a month from five sampling sites within IFE. The samples are analysed directly for gamma emitters. All the rainwater is collected to quarterly samples and analysed for ^{3}H, ^{90}Sr, ^{238}Pu and $^{239,240}Pu$.

Figure 1
Rainwater collector

Grass:
Grass samples are collected at five sampling sites within IFE and six sampling sites outside IFE. The samples are collected twice before the summer holidays and twice after. All grass inside a ¼ m^2 frame is gathered, cut just above the soil layer.

Each sample (50-70 g fresh grass) is analysed directly for gamma emitters, mainly ^{131}I and ^{137}Cs. Samples gathered within IFE, and samples gathered outside IFE, are combined to two pooled samples, respectively. These are analysed radiochemically for ^{238}Pu, $^{239,240}Pu$ and ^{90}Sr.

Milk:
Milk is collected once a week from two farms near IFE. The samples are analysed for gamma emitters. The milk is also pooled to quarterly samples which are analysed for ^{90}Sr.

Agricultural samples:
Agricultural samples (grain) are collected every autumn from farms near IFE. The grain samples are first analysed for gamma emitters and then for ^{90}Sr, ^{238}Pu and $^{239,240}Pu$.

Surveillance of discharges to Nitelva River and recipient control

The low level liquid radioactive waste is analysed for radionuclides before discharge to Nitelva River through an underground pipeline (NALFA). The water and the sediments (if any) inside the pipeline system are also analysed for various radionuclides. Radioactivity levels in the Nitelva River are monitored by analysing samples of water, sediment, aquatic plants and fish (Table 2).

Table 2. Surveillance of discharges to Nitelva river

Sample	Number of sampling sites	Sampling sites
Water	6	Approx. 2,7 km upstream of the discharge point to approx 13 km downstream of the discharge point
Sediment	6	Approx. 2,7 km upstream of the discharge point to approx 13 km downstream of the discharge point
Aquatic plants	1	Aprox 0,5 km downstream of the discharge point
Fish	1	1.2 km downstream of the discharge point
NALFA-waste pipe	9 13	Within IFE´s area Outside IFE´s area, from IFE to the discharge point

Low level liquid radioactive waste from IFE´s activities:
The liquid waste water is collected in large tanks at IFE before discharge through the NALFA pipeline. The water in the tanks is sampled and analysed before discharge. Based on the gamma spectrometric and ^{3}H results, the water is either cleared for discharge or held back for further decay. The waste water is also analysed for ^{90}Sr, ^{238}Pu, $^{239,240}Pu$ and ^{241}Am.

Surveillance of the NALFA pipeline:
As a part of the surveillance programme of the NALFA pipeline, sediment (if any) and water samples are annually gathered from all manholes between IFE and the discharge point. The sediments are analysed for gamma emitters ^{238}Pu and $^{239,240}Pu$, and the water is analysed for gamma emitters, ^{90}Sr and ^{3}H.

River water:
Water samples are collected three times during the summer, from six different sites in Nitelva River. The samples are analysed for gamma emitters, ^{90}Sr, ^{238}Pu and $^{239,240}Pu$.

Fish and aquatic plants:
Fish and aquatic plants in the Nitelva River are collected twice during the summer. Only the three most common fish species; perch, pike and bream are sampled. The aquatic plant collected is *Potamogeton perfoliatus*. The samples are analysed for gamma emitters, ^{3}H, ^{90}Sr, ^{238}Pu and $^{239,240}Pu$.

Sediment samples:
The sediment samples (cores) are collected from six different sites in Nitelva once a year. One of the sites is approx 2,7 km upstream the discharge point and serves as a background site. The samples are analysed for gamma emitters, ^{90}Sr, ^{238}Pu, $^{239,240}Pu$ and uranium.

Figure 2
Equipment for sedimentsampling

Surveillance of liquid releases from KLDRA-Himdalen

The drainage water is collected annually from three different drainage systems. The samples are analysed for gamma emitters, ^{3}H, ^{90}Sr, ^{238}Pu, $^{239,240}Pu$.

Surface water collected annually from two small lakes nearby is analysed for gamma emitters. A more detailed sampling programme in the environment of KLDRA-Himdalen is carried out every 5th year.

Conclusion

The detailed surveillance programme ensures that the monitoring of IFE´s discharges, both within IFE and in the surrounding areas, fulfills the requirements of the authorities. It is also important in mainaining IFE´s expertise in radiological monitoring and radiochemical analysis. The results show that the levels of discharged radionuclides in the surrounding areas are low, mostly indistinguishable from global and Chernobyl fallout. Occasionally higher than background levels of ^{131}I are measured in grass and precipitation samples collected within IFE.

References

IFE (2004) Quality Assurance Manual for the Environmental Monitoring Section, Health and Safety Department, (in Norwegian) 3rd edition

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Academic and non-academic education in ionising radiation protection in Sweden

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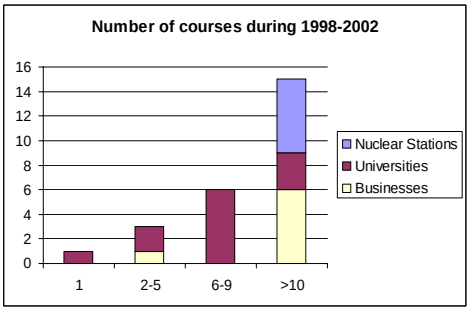
Education in Ionising Radiation Protection (RP) in Sweden

- Objective 1
  - Trace existent educational RP programmes. (sponsored by SSI)
  - executed by B-A Jönsson, B. Roos and C. Samuelsson
- Objective 2
  - Teach custom officers gamma ionisation metrology (sponsored by SSI with funding from KBM)
  - executed by C. Östlund and C. Samuelsson

Finding courses in radiation protection

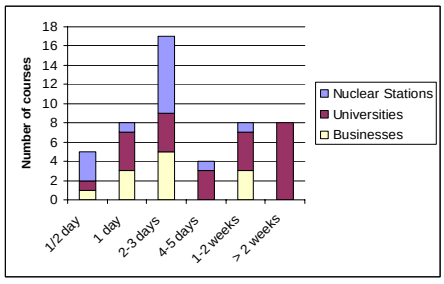
- A query was sent out by email in May 2003
- The query could be filled in electronically
- 66 queries were mailed and we received 30 answers, 28 offered relevant courses
- More than 1 person received the query at large organisations
- The responders were divided into 3 categories, Nuclear power, Other businesses, Universities.
- As a complement to the query the new Hospital Physicist curriculum was scrutinized
- We aimed at covering statistics for the 2001-2003 time period.

Examples of Query results



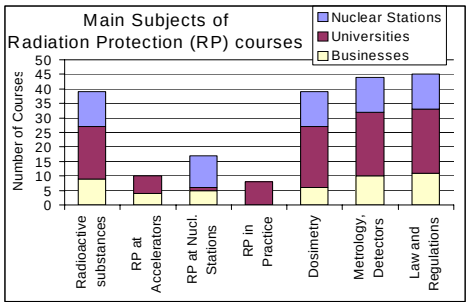
Examples of Query results

The duration of the radiation protection part of a course



Examples of Query results

(A single course can cover several subjects)



Giver	Course ID	Duration	Disciples	Year
Stensand	Strålskydd grund tekniker katetgori C	1-2w	10	2003
Stensand	Strålskydd Kategori B	4-5d	10	2003
Stensand	Strålskydd grund	2-3d	7,5	2003
Elajo	Strålskyddskurs för nya C-arbetare	1-2w	5	2003
Elajo	För strålskyddstekniker i kategori B	4-5d		
ISS	Hjälptekniker	2-3d	20	2002
ISS	Strålskydd C	1-2w	10	2003
ISS	Strålskydd B	1-2w	5	2002
Forsmarks AB	Strålskyddsteknik grunder	2-3d	17,5	2003
Forsmarks AB	Strålskyddsteknik grunder (även stationstekniker)	2-3d	17,5	2003
Forsmarks AB	Strålskyddsteknik fördjupning	2-3d	12,5	2003
Oskarshamn	Strålskyddsteknik grunder	2-3d	10	2003
Oskarshamn	Strålskyddsteknik, mätuppgiftspersonal	2-3d	10	2003
Ringhals	Strålskyddsteknik grunder	2-3d	15	2003
Ringhals	Strålskydd repetition	1/2d	15	2003
Studsvik	Strålskyddskurs grunder	2-3d	20	2002
Studsvik	Strålskyddskurs, rep	1d	60	2003
Studsvik	Strålskydd special	1/2d	10	2003

Mean number of Disciples, Data from KSU not included.

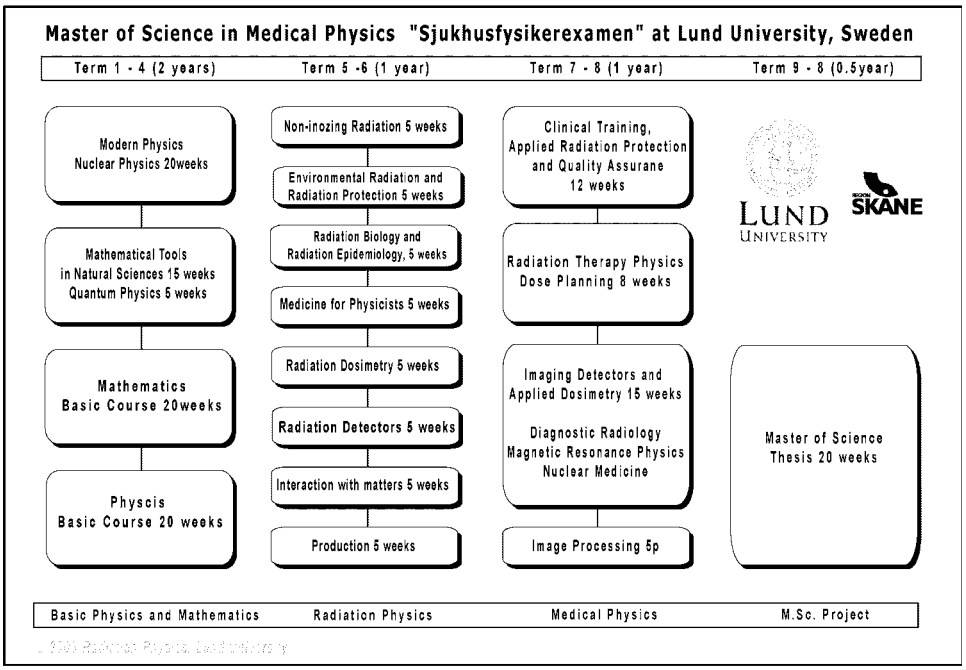
The Academic Level

- About 10 academic institutions/departments in Sweden provide courses containing significant RP elements on the academic level
- The universities in Gothenburg, Lund, Stockholm and Umeå provide a Master of Science in Medical Physics (Legitimerad Sjukhusfysikerexamen)
- This MS in Medical Physics comprises 180 p (weeks) and is a health profession registered by the state
- Academic courses awarding ligitimatization/ authorization of RP specialist outside the medical field do not exist in Sweden

No. of Degrees 2003-2005

Department University	Degree	Weeks	No. Of degrees (LADOK) 2003	2004	2005
Med Rad Phys. Stockholm	MS in Rad Phys.	160	?	?	?
	MS in Med Phys.	180	3	1	14?
Rad Phys. Gothenburg	MS in Rad Phys.	160	3	?	?
	MS in Med Phys.	180	3	?	?
Med Rad Phys. Lund	MS in Rad Phys.	160	na	na	na
	MS in Med Phys.	180	4	7	11?

(Data for the Master of Engineering degree in Umeå was not available)



A 4-week course for custom officers

- Mobile boarder control of illicite trafficing with radioactive sources and nuclear devices
- 3 teams with cars fully equipped with gamma spectrometry and neutron detectors
- 4-5 persons in each team with no or little experience of ionising radiation beforehand
- April/May 2005, 2 weeks of theory and exercises with handheld dosemeters and spectrometers, but without cars
- 2 weeks in October 2005 in-situ with fully equipped cars and safety equipments

What do you need in order to teach laymen perform and interpret ionisation spectrometry?

1. Well motivated laymen
2. Regognition of and respect for the laymen level
3. 2 weeks of theory and experiments
4. Start with everyday physics (the custom officers started with arm wrestling!)
5. No radiation until concepts like *force, binding energy, matter, atoms*, etc are known and practised
6. Interactive and feedback type of pedagogy
7. An appropriatly written textbook
8. An appropriate assortment of modern detectors for everybody to practise on

Acknowledgement

The authors thank SSI for funding the enquiry on radiation protection courses and Birgitta Roos for helping us with administrating the elecronic query.

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The Medical Radiation Physicist Programmes in Sweden: Lund University www.radfys.lu.se; Göteborg University www.radfys.gu.se; Stockholm University www.ki.se/onkpat/radfys/index.html; Umeå University www.umu.se/radsci/radiofysik/.

A student's thoughts and visions about radiation protection

U. Estenberg

Department for Radiophysics, University of Göteborg, Sweden

A student's thoughts and visions about radiation protection

Ulrika Estenberg

Department for Radiophysics, University of Göteborg

Lecture

Team work

Laboratory work

Report

Differences

- Large classes
- Anonymous
- Short learning process
- Several courses

- Small groups
- Personal guidance
- Vocational
- Single courses

Future visions

- Advanced diagnostics
- Advanced radiation therapy
- Disposal of radioactive waste

Technologies to explore gamma radiation influence on structurally depended exoemission properties of bone

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Abstract: This article deals with the estimation of inserted defects in bone material by gamma radiation used for radiation therapy. Exoelectron emission (EE) phenomenon underlies a new electron spectroscopy to explore alteration of the electronic structurally dependence properties of bone material. The influence of gamma radiation with different energies on the bone structure had been discussed.

Introduction

Gamma radiation is used to treat carcinogenic diseases. Gamma radiation used for therapy has the energy of photons varying from 1-20 MeV. These photons do not kill the cancer cells only, but they also have effects on the structures of bones. Structure alters electron density of states that, in time, influences biomedical reactions on bone life condition. Due to this reason, biochemical reactions between the bone and the surrounding cells targeted to remodel tissue could be influenced *E. F. Eriksen (1992)*. A bone surface, where the cells are coming to be adhered *J. M. Schakenraad (1996)* with the aim to supply biochemical reactions *D. W. Demster (2000)*, provides the greatest impact.

Radiation stimulated processes are initiated due to generation of single imperfections such as vacancies, interstitial atoms, radicals, etc. in a very short time *Yu. Dekhtyar (1996)*. According to *Yu. Dekhtyar (1997)*, inserted defects facilitate diffusion-controlled reaction developed in the long-time scale to move a solid material into the state of thermodynamically equilibrium. The exoemission analysis is widely used for irradiated semiconductors as siliceous, carbon plastic-reinforced material and natural bone composites. In all these cases, the total exoemission-related charge is directly proportional to the number of radiation-inserted imperfections and possessed wave-type regularity on dependence of the exposure *Yu. Dekhtyar, (2000).*

The interest of this study is to develop EE measurement technique to be in use for investigating gamma radiated bones, as well as to perform analysis of gamma radiation effect on the bone structure.

Material and methods

To investigate the gamma radiation effects, the bovine bone had been chosen as a prototype for the human ones. The bovine bone has components and structure similar with human bone according to *J.L. Katz (1995)*. The bovine tibia bone of slithered 1.5 years old cow was used as a material for the experiments. The bovine bone was bought after

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two days the animal was slithered. Then, it was cleaned mechanically from muscles, fat and bone marrow. Specimens were cut transversally to the bone longitudinal axis using the stainless steel surgical saw or facing tool and have the same physical parameters: thickness (A) of 0.2 cm, surface area (B) of 0.36 cm^2, volume of 0.36cm^3 and mass density of 0.9g/cm^3.

Measurements were performed using the exoemission technique which was described in details by *Yu. D. Dekhtyar (1981).* Exoemission spectrometer consists of a vacuum system, a complex of electronic devices for the EE intensity registration, as well as the record of the received information blocks of a photo- and thermostimulation. The photon energy, having a very strong impact on the current, used for photostimulation was equal to 5.6 eV. The chosen heating rate was equal to 10^{o}C/min according to recommendation of *T. Bogucharska (2000).*

The bone material has specific EE measurement technology. Exoemission measurements should be carried out after the 10th day of storage of the samples at room conditions and the measurements should be performed in a time less than five days after gamma irradiation. The heating cycle has an effect on the exoemission response and the measurements cannot be provided after the 1st heating cycle. Heating of the specimens till +200^{o}C during EE measurement changes the bone emission properties in the irreversible manner. To perform the calibration of the specimens before and after irradiation, the EE current (I) was measured using only photostimulation.

To evaluate the final value of imperfections inserted by gamma radiation in bone material, the photo- and thermo-stimulation were applied simultaneously. Square of the area under the EE graph is directly proportional to the total number of electrons (Q) emitted during EE measurement:

$$Q = \int_{T_0}^{T_f} I(T)dT$$

where T_0 and T_f are the initial and the final temperatures of EE spectrum, respectively. The total number of electrons (Q), emitted during EE measurement is directly proportional to the concentration of the inserted by gamma radiation defects *Yu. D. Dekhtyar (1981).* Standard uncertainty of exoemission measurement was 4% with the level of significance of 95%.

Gamma irradiation had been performed using the linear accelerator SL75-10 placed at the Latvian Ontological Centre. The irradiation had been performed using the gamma radiation photons with the energies of 8 and 18 MeV respectively. The chosen absorbed dose rates $\dot{D}$ were 2 and 10 Gy/min. Gamma radiation absorbed doses (D) applied on the bone specimens varied till 10 Gy. The bone specimens were irradiated during the same day.

Table 1. Total number of experiments performed irradiation.

Irradation mode	Number of experiments	E, Mev	D, Gy	$\dot{D}$, Gy/min
1	69	6	2,4,6,8,10	2,10
2	69	18	2,4,6,8,10	2, 10

Results and discussion

The EE measurement of the irradiated and non-irradiated bone materials' is shown in Figure 1. Due to the annealing of the specimens, the reflection of the imperfections generated by gamma radiation is seen.

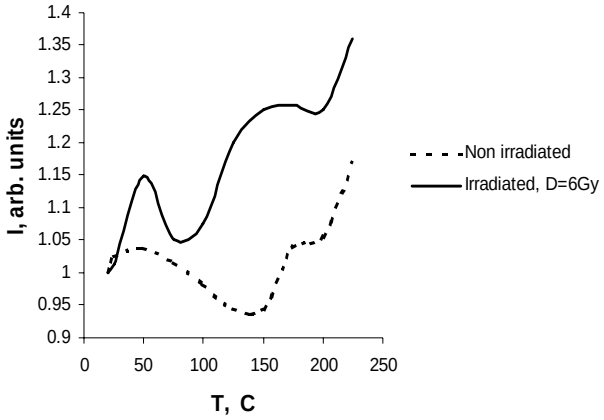

Figure 1. Typical EE spectra of bone before and after photon irradiation.

Figure 1a shows the structural changes inserted in the bone material by gamma photons radiation with energy equal to 6 MeV described by total emitted charge Q. The experiments had been performed with different absorbed dose rates 2 Gy/min and 10 Gy/min. The applied absorbed dose range varied until 6Gy. According to the achieved results, the non-linear growing of imperfections in bone samples is clearly seen. In case of magnitude, gamma photon energy equals to 6MeV with different absorbed dose rates does not affect the number of radiation inserted imperfections.

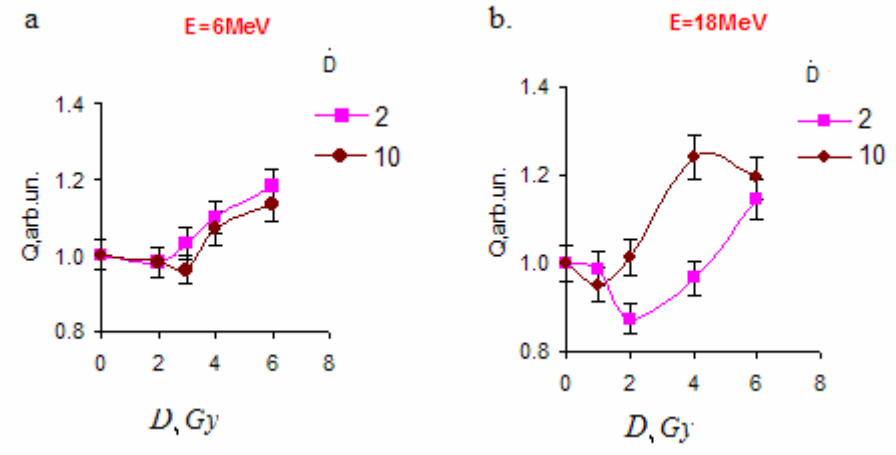

Figure. 2. Absorbed dose rate effect on total emmited charge Q depending on absorbed dose D for gamma photon with energy: a. E=6 MeV; b. E =18MeV

On another side, figure 2b shows the structural changes inserted in the bone material by gamma photons with energy of E=18MeV. It is clearly seen that the same gamma radiation dose rates has influence on generation of imperfection in bone material.

Irradiation with different photon energies demonstrated non-linear behavior of total emitted charge Q after constantly increasing absorbed dose rate. The tendency of peaks magnification affected by higher dose rate is found for energy equal to 18MeV.

Conclusion

One can assume that the bone EE response is not linear on the dose rate and probably is connected with interaction of radiation induced centers responsible for EE process.

EE structural changes have varied with energy variety of doses used in radiotherapy.

Acknowledgement

Special thanks to the Nordic Society for Radiation Protection (NSFS) for granting me the scholarship that is equal to the registration fees as support to attend the XIV NSFS meeting/conference in Rättvik, Sweden.

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Response time measurements for hand-held dose rate measurements

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Introduction

A dose rate meter's response time is the time it takes for the instrument to display the "correct" value after a sudden change of dose rate. It is an important factor to know when you are going to perform dose rate measurements or search for orphan sources.

FOI NBC-defence has developed a method to do response time measurements for hand held dose rate instruments.

The camera samples 10 display readings per second. The radiation times in our tests are 60 seconds. The start and end point of the radiation is when the sensor detects that the radioactive source has reached or left the radiation position.

By image interpretation then the registrations are converted into measuring values, which can be presented as graphs or used in calculations.

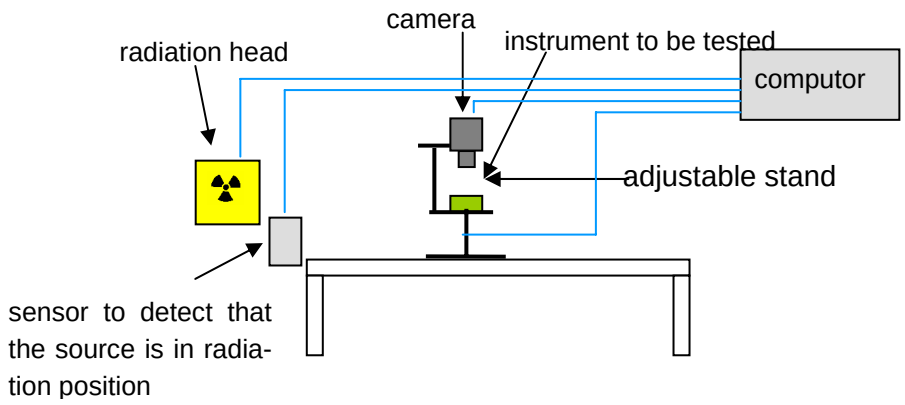

Figure 1. Schematic description of measurement system.

Results

We have so far made time response measurements for five different dose rate meters. The measurements are done at four reference dose rates, Table 1.

Table 1. Reference dose rates

Dose rate nr	Ref value(mSv/h)
1	110
2	0.94
3	0.52
4	0.10

As an example the results for one of instruments tested are shown in Figure 2.

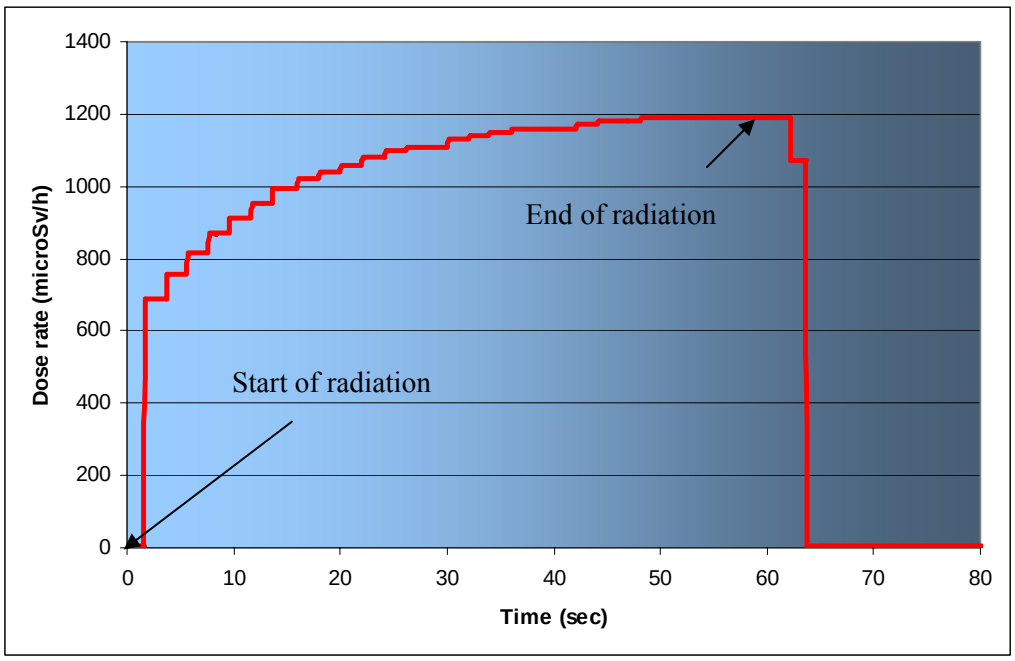

Figure 2. Example of measured response for one of the instrument.

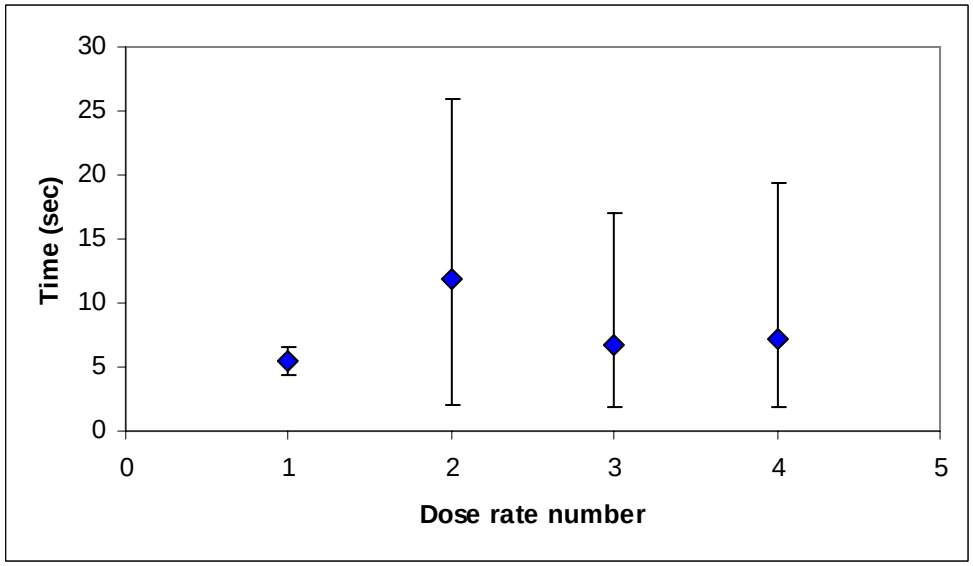

Figure 3. Instrument response at four different dose rates for one instrument type. The figure shows the mean value time for the instrument to reach 90% of the reading the instrument will show when it stabilises. Ten instruments and five measurements for each dose rate were used in the test. In the figure the maximum and minimum times also are shown. The dose rate numbers are according to Table 1.

Our tests show, as can be seen from Figure 3, that there are great variations in response times for different dose rates, and for single instruments within each instrument type. Not shown in this poster we have noticed that there are great variations between different instrument types as well.

Analysis intercomparison of lead-210 in aerosol filters

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Analysis Intercomparison of Lead-210 in Aerosol Filters

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INTRODUCTION

Finnish Meteorological Institute (FMI), STUK - Radiation and Nuclear Safety Authority, and Laboratory of Radiochemistry, University of Helsinki (UHRAD), made an intercomparison of lead-210 analysis of aerosol filters. Two high-volume aerosol samples with an air volume of ca. 3500 m³ were collected onto Munktell MGA glass fibre filters in Ivalo, Northern Finland, in December 1998. After the sampling the filters were stored to allow bismuth-210 and polonium-210, the beta and alpha emitting daughter nuclides of lead-210, to grow in.

EXPERIMENTAL METHODS

Finnish Meteorological Institute measured the lead-210 content of the aerosol filters by counting the alpha particles of the in-grown polonium-210 instrumentally directly from the filter surface. An automatic alpha/beta analyser equipped with gas-flow proportional counters was used in the study. After one measurement in 1999 and 5 consecutive measurements in 2002 the filters were cut to two halves. One half of each filter was sent to STUK - Radiation and Nuclear Safety Authority and the other half to Laboratory of Radiochemistry.

STUK - Radiation and Nuclear Safety Authority used three different methods to assay lead-210 in the filter fractions, low-energy semiconductor gamma spectrometric assay of lead-210, wet digestion followed by beta counting of bismuth-210 with a liquid scintillation spectrometer, and spontaneous deposition of polonium-210 onto silver disks followed by alpha spectrometry.

Laboratory of Radiochemistry analysed their filter fractions with wet digestion and spontaneous deposition of polonium-210 onto silver disks followed by alpha spectrometry.

Figure 1. High-volume filter sampler.

Figure 2. Automatic alpha/beta analyser at the FMI.

Figure 3. Liquid scintillation spectrometer.

RESULTS

The differences between results obtained with the different analysis methods were not consistent between the two filter samples. The spread of observed activity concentrations using different methods emphasizes the need for continuation of intercomparison exercises on both a national and an international level.

Sample 1		**Sample 2**	
Measurement	Lead-210 act. conc. µBq/m³ ± 1 σ	Measurement	Lead-210 act. conc. µBq/m³ ± 1 σ
UHRAD	123 ± 4	UHRAD	161 ± 5
FMI 1999	110 ± 10	FMI 1999	156 ± 11
FMI 2002	97 ± 8	FMI 2002	151 ± 7
FMI 2002	107 ± 6	FMI 2002	154 ± 5
FMI 2002	101 ± 6	FMI 2002	150 ± 5
FMI 2002	92 ± 8	FMI 2002	140 ± 7
FMI 2002	107 ± 8	FMI 2002	151 ± 7
STUK gamma	101 ± 13	STUK gamma	223 ± 29
STUK alpha	131 ± 7	STUK alpha	180 ± 9
STUK beta	142 ± 14	STUK beta	196 ± 20

Session IV: Emergencies

Dose contributions from malicious radionuclide dispersion in an urban area

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Abstract: If a 'dirty bomb' radioactivity dispersion device were detonated in a city centre, the radiological consequences could be anything from trivial to severe, depending on factors like the radionuclides dispersed, the amount of radioactive matter, and the type of explosion. To shed light on the possible impacts on humans, the relative contributions of the various relevant dose pathways are estimated for a 'dirty bomb' example scenario.

Introduction

The possibility of terrorist attacks involving devices for malicious dispersion of harmful substances constitutes a threat to society. So-called 'dirty bombs' for dispersal of radioactive matter using conventional explosives could relatively simply be constructed by terrorists that can gain access to radioactive sources. Strong 'orphaned' sources, which no longer serve a purpose and are not kept track of, exist, e.g., in the former Soviet Union, and may find their way into illegal trading. ^{90}Sr is one of the more obvious candidates, as it has been widely used in lighthouses, beacons, and other unmanned facilities. ^{90}Sr is also advantageous over most in that it is a pure beta emitter. Thus only little shielding material is needed to enable safe handling of the bomb prior to detonation. This paper considers different contributions to dose that might be received by persons living in a city area, due to airborne ^{90}Sr contamination from a 'dirty bomb' attack.

Generic methodology

A number of assumptions need to be made to describe any plausible 'dirty bomb' scenario. An important factor in determining the distribution in the environment of the dispersed contaminants is the size distribution of the contaminant particles generated by the blast. The information available from the open literature to describe this particle size distribution is very limited, and not directly related to 'dirty bombs'. However, in January 1968, when an aeroplane carrying 4 nuclear weapons exploded about 12 km away from Thule Air Base in Greenland, the process of dispersion of radioactive material was essentially consistent with that of a 'dirty bomb'. Particle size distributions were here reported to be lognormal with a mean of 2 μm and a log standard deviation of one (Danish Atomic Energy Commission, 1970). This means that only 1.3 % of the particles were larger than ca. 18 μm, but these carried nearly 80 % of the activity (Eriksson, 2002). This size distribution pattern is in-line with dust observations made in connection with detonation of high explosives in/on various soils (Pinnick et al., 1983). Investigations of particle size distributions (as reported by differential mass) here revealed a distinct peak at a mean particle radius of ca. 7 μm, but a more limited number of particles in the ca. 100 μm range were also observed. The tests were conducted in areas of soils with very different textural features, ranging from desert sand to clayey soil. However, no obvious connection was observed between soil particle size distributions and dust distributions. It was on that basis believed that a combination of disaggregation, coagulation and settling processes transform widely differing target materials, on which the bomb impacts, into dust of common size distribution (Pinnick et al., 1983). The limited number of large hot particles in the 100 μm range generated by the blast would, due to strong gravitational settling, not remain airborne for long time. In contrast, small particles, on which gravity

would have considerably less influence, would be airborne over much longer periods of time and could reach correspondingly greater distances from the release point. Naturally, the size of the affected area will depend on the magnitude of the blast, which determines the effective 'release' height. The chemical form of the ^{90}Sr source can be important in the context of solubility. Strontium compounds range from the insoluble to readily soluble. For this example, it is assumed that the strontium particles are not readily soluble. It would be most likely that a 'dirty bomb' would be detonated in dry weather, as this would give the greatest dispersion and the most diverse contributions to dose.

An estimation of the dose contributions received through different pathways is presented in the following sections. Internal doses from inhalation of resuspended dust as well as external doses from the passing cloud have been left out since they would be likely to be of little significance (see, e.g., Andersson et al., 2004).

Results for external dose from contaminated outdoor surfaces

The dose rate that is received at the depth of the basal layer of the epidermis of human skin from a large ground surface uniformly contaminated with ^{90}Sr has been estimated by Eckerman & Ryman (1993) to be of the order of 4 10^{-13} Sv h^{-1} per Bq m^{-2}. Doses to inner organs would typically be at least 3 orders of magnitude lower, and thus be likely to be of very little significance. A requirement to reach as high dose rates as this would be that the contamination lies on the very surface of the ground. If it is 1 cm down in soil, as would be expected rather shortly after an airborne contamination, the shielding effect is so great that the dose rate to the skin would be about 3 orders of magnitude lower. This dose rate contribution may thus well be reduced to a negligible value at the time when the first heavy rain shower arises. This is here assumed to happen 30 days after the deposition. On average, it seems reasonable to assume that people spend some 10 % of the time outdoors (Andersson et al., 2004). This would give a dose contribution estimate of 10 % * 4 10^{-13} Sv h^{-1} per Bq m^{-2} * 30 days * 24 h day^{-1} ~ 3 10^{-11} Sv per Bq m^{-2}. Contaminant particles will naturally also deposit on other outdoor surfaces. As gravitational settling will dominate for relatively large supermicroneous particles, the highest contaminant concentrations will be expected on horizontal surfaces. Large and rather insoluble particles will much more rapidly be removed from streets by natural weathering processes than will smaller particles. This is reflected in the high decontamination factor of 50, obtained by Clark and Cobbin (1964) when hosing water on a street contaminated by particles in the 44-100 μm range. Routine street cleaning and natural weathering processes would therefore be likely to rapidly result in very low street contamination levels, as the particles are assumed not readily to go into solution.

Results for external dose from contaminated indoor surfaces

It is here assumed that windows and doors are closed during the contaminating episode. For supermicroneous particles it is not unreasonable to assume that nearly all the deposited indoor contaminants will be distributed on the horizontal surfaces, primarily the floor. It can be shown (Andersson et al., 2004) that the deposition, D_i, per unit indoor area can be estimated from the formula: $D_i \sim (D_0 / v_{d,0})\, h\, f\, \lambda_d\, \lambda_v / (\lambda_d + \lambda_v)$, where D_0 is the corresponding deposition on an outdoor reference surface (e.g., a cut lawn), $v_{d,0}$ is the deposition velocity to that reference surface, h is the room height (typically some 2.5 m), f is the filtering factor, λ_d is the rate coefficient of deposition indoors and λ_v is the rate coefficient of ventilation of the dwelling. For particles in the relevant size range (ca. 5-100 μm), λ_d would be high compared with λ_v, so that $D_i / D_0 \sim h\, f\, \lambda_v / v_{d,0}$. With typical values for ca. 5-10 μm particles (Long et al., 2001) this gives: $D_i / D_0 \sim 2.5$ m*0.5*0.5 h^{-1} / 10^{-2} m s^{-1} ~ 0.02. A likely half-life of the natural removal process of the smallest relevant particles on the floor would be of the order of six months (Andersson et al.,

2004). A conservative estimate of the dose rate can be obtained using the same dose conversion factor as for outdoor horizontal surfaces. With the above assumptions on natural removal, the total external dose from contaminated 5-10 μm particles deposited on indoor surfaces (taking into account the fractions of time spent indoors and outdoors) could under the given circumstances be estimated to be of the order of (200 d/30 d)* 0.02 * 0.9/0.1 = 1.2 times that from the corresponding size fraction of contaminants deposited on outdoor surfaces. For the larger particles, which will carry by far the most of the contamination, the relationship between indoor and outdoor deposition will be very much smaller since f will decline (Long et al., 2001), and $v_{d,0}$ will increase. If it is assumed that external dose from indoor contaminants is largely governed by perhaps 10 % of the activity associated with particles that can adequately be described by the characteristics of 5-10 μm particles, a rough estimate of this dose would amount to some 10 % * 3 10^{-11} Sv per Bq m^{-2} * 1.2 ~ 4 10^{-12} Sv per Bq m^{-2}. It should be noted that as even thin fabric shields well against beta radiation, the most critical situations would be those where unshielded skin comes into direct or close contact with a contaminated surface.

Results for external dose from deposition on human skin

The most critical contamination of humans would occur to freely skin of persons who are outdoors during virtually the entire period of deposition, so this is assumed. If people are indoors with windows and doors closed, the deposition on humans of 5-10 μm particles will typically be at least one order of magnitude less, and even less for larger particles. Deposition velocities of 5-10 μm particles to human skin are high: of the same order of magnitude as that to the much rougher grassed reference surface. Thorough washing will be rather efficient in removing these particles from skin, and even more efficient in removing larger particles. However, if nothing active is done to remove the particles (e.g., when no warning is given), the natural removal of 5-10 μm particles would occur with a half-life of max. ca. 1 day, and for larger particles, the half-life might well be considerably less (Andersson et al., 2004). Current knowledge is inadequate to allow estimation of the length of time that say 50 μm particles could remain on the skin, and thus the impact that these could have on dose. However, as such particles would be expected to have very high deposition velocity on skin, further experimental investigations are merited. If we look at the perhaps 10 % of the total airborne activity that is associated with particles with an AMAD in the range of 10 μm, it can be found (Andersson et al., 2004) that with the above assumptions, the corresponding dose would amount to 10 % * 8 10^{-10} Sv per Bq m^{-2} ~ 10^{-10} Sv per Bq m^{-2} on the grassed reference surface. This alone is approximately three times the external dose contribution from outdoor contaminated surfaces.

Results for dose from inhalation during plume passage

Inhalation doses are essentially found by multiplying *time-integrated* contaminant air concentrations by an ICRP dose conversion factor and an inhalation rate. It may well be that large particles represent 80 % of the activity or even more, but they would clearly have considerably less significance for the time-integrated air concentration, as they would deposit quickly. Moreover, it has been stated by the ICRP (1993) that inhaled 'particles larger than 10 μm would be cleared rapidly by ciliary action'. According to the same ICRP publication, the fractional deposition in the bronchial region is several orders of magnitudes less for 100 μm particles than for 10 μm particles. Assume again that 10 % of the total airborne activity is associated with particles with an AMAD in the size range of 10 μm or less. The time-integrated air concentration outdoors, C_o^{int}, that will lead to a deposition of 1 Bq m^{-2} on an outdoor grassed reference surface can be found by dividing the 1 Bq m^{-2} by the deposition velocity to the reference surface (for this particle size some 10^{-2} m s^{-1}). This gives 100 Bq s m^{-3}. By multiplication of this time-integrated

air concentration with an inhalation rate ($3.3\ 10^{-4}$ m^3s^{-1}; ICRP, 1993) and a factor of 0.1 (assuming only some 10 % of the airborne contamination is inhalable), the total inhaled amount of contamination becomes 100 Bq s m^{-3} * 0.1 * $3.3\ 10^{-4}$ m^3s^{-1} = $3.3\ 10^{-3}$ Bq. According to ICRP, the committed dose by inhalation of 1 Bq of ^{90}Sr would be $1.6\ 10^{-7}$ Sv (class S). The inhalation thus here gives a dose of $3.3\ 10^{-3}$ Bq * $1.6\ 10^{-7}$ Sv per Bq = $5\ 10^{-10}$ Sv per Bq m^{-2} on the reference surface. This is higher than the other dose contributions estimated above. However, it should be stressed that this dose would be much smaller if people stayed indoors during the time period of elevated air concentrations, and if the aerosol had been more readily soluble.

Discussion and conclusions

Through an example, methodologies are demonstrated that can be applied to analyse which dose contributions from a 'dirty bomb' attack might dominate under different conditions. It is clear from the above that there are gaps in the current knowledge, due to which a *full* description of the dose aspects of a 'dirty bomb' event is currently not possible. These should be addressed by further research. It is evident that the current European preparedness models are not equipped for adequate estimation of doses from 'dirty bombs', as many parameters that apply to a 'dirty bomb' incident are very different from those applied in 'traditional' models targeted on nuclear power plant accidents. Nevertheless, the incorporation of currently known parameters of relevance to 'dirty bombs' in a comprehensive model system would be highly advantageous, as this could facilitate sensitivity analyses by parameter variation, to pinpoint weaknesses that merit further research. The above calculation examples demonstrate that it is possible to build up a parameter database to facilitate estimation of doses from a 'dirty bomb' incident.

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Probabilistic derivation of dose rate action levels for urgent countermeasures

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Abstract: A probabilistic approach to derive action levels for emergency countermeasures in terms of stationary gamma-radiation measurements is discussed and simple means for calculation of action levels presented. The method is applied to a nuclear accident situation with long-range atmospheric dispersion of radionuclides where dose rate action levels are estimated for sheltering and foodstuff restrictions.

Introduction

Automatic gamma-monitoring systems play an important role in nuclear emergency preparedness. Local monitoring at a nuclear power plant may provide early warning about accidental releases from the facility while countrywide monitoring networks may detect releases from foreign or domestic sources. In case of a nuclear accident, however, data from gamma monitoring networks may also be used to assess the severity of the accident and the need to introduce intervention measures for the protection of the public.

The benefit of a countermeasure is usually quantified in terms of the avertable dose; the countermeasure will be justified when the expected avertable dose is larger than an optimized intervention level (IL). Generic ILs for urgent countermeasures like sheltering, evacuation, iodine prophylaxis and restrictions on the use of foodstuffs are recommended by international bodies [1], and implemented in national legislation. The avertable dose, however, is not a measurable quantity and intervention criteria, especially for urgent countermeasures, are better defined directly in terms of the gamma dose rate or similar measurable quantities.

Action levels in a probabilistic setting

To be effective, emergency countermeasures should be introduced at a time when information on the scale of the accident is very uncertain, and due account must be taken to the variability and uncertainty associated with the accident release scenarios, the environmental transport of radionuclides, the effectiveness of countermeasures and the detection of gamma radiation from a passing radioactive plume. No simple one-to-one correspondence exists between the measured gamma radiation and the avertable dose. Rather, the avertable dose should be described as a stochastic variable, allowing only for probabilistic assessment of the benefit of a countermeasure.

In the statistical description there is no need to distinguish between variability, or heterogeneity, in exposures that may or may not be known at the time of the accident, and genuine uncertainty, *e.g.* associated with the release characteristics and with forecasting atmospheric transport of radionuclides. The outcome of a protective action can be quantified in terms of a conditional probability density function (pdf) of avertable dose, $p(\Delta E \mid d)$, where d is the measured dose rate. The pdf may be estimated based on as-

sumptions on the dispersion and exposure to radionuclides from the accident and the nature of the contamination. Note that the protective action, hence ΔE, in general applies to different locations than the position of the radiation measurements.

Two levels of dose rate can be introduced as basis for decisions on specific countermeasures. At a lower, dose rate action level, d_{AL}, there is a small but finite chance, say r_{AL}, that the avertable dose exceeds the intervention level,

$$\Pr\left(\Delta E > \Delta E_{IL} \mid d_{AL}\right) = \int_{\Delta E_{IL}}^{\infty} d\left(\Delta E\right) p\left(\Delta E \mid d_{AL}\right) = r_{AL} \,. \tag{1}$$

One may identify the action level d_{AL} with possible introduction of precautionary countermeasures. A cautious approach for the introduction of a countermeasure corresponds to choosing a small value of r_{AL}, *e.g.* r_{AL} = 5%. At a larger, operational intervention level (OIL), d_{OIL}, countermeasures may be introduced automatically. The OIL may naturally be defined by requiring the expectation of the avertable dose to be equal to the intervention level for the protective action [2],

$$\Delta E_{IL} = E\left(\Delta E \mid d = d_{OIL}\right) = \int_{0}^{\infty} d\left(\Delta E\right) \Delta E \; p\left(\Delta E \mid d_{OIL}\right) . \tag{2}$$

In Fig. 1 the two levels are indicated; it is implicitly assumed that the dose rate action level is smaller than the operational intervention level. The relation between the two levels, however, depends on the details of the pdf and the presumption $d_{AL} < d_{OIL}$ is only guarantied for a sufficiently small probability r_{AL}.

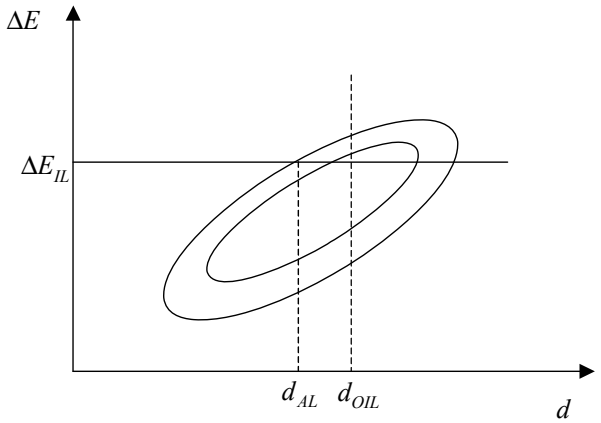

Figure 1. Schematic drawing of joint probability density function $p\left(\Delta E, d\right)$.

Because modeling uncertainty may be considerable, especially for long-range atmospheric transport, direct application of the definitions (1-2) may not be practicable. On the other hand, for some urgent countermeasures, e.g. sheltering, gamma dose rates and avertable doses may be strongly correlated, i.e. the variability of both the release and the atmospheric transport will have a similar effect on the dose rates and the avertable doses. Hence, estimation of the ratio of dose rate to avertable dose, $d/\Delta E$, will be less uncertain and also, this ratio will be much less sensitive to variation of the transport distance from the release point. Eq. (1) can be rewritten as

$$r_{AL} = \Pr\left(\Delta E > \Delta E_{IL} \mid d_{AL}\right) = \Pr\left(z < d_{AL} \,\middle|\, d_{AL}\right) = \int_{0}^{d_{AL}} dz \; p\left(z \mid d_{AL}\right); \quad z = \frac{d}{\Delta E} \Delta E_{IL} \tag{3}$$

In case of a strong correlation between d and ΔE we may approximate $p(z \mid d_{AL})$ by the unconstrained pdf [3], *i.e.*,

$$r_{AL} \approx \int_0^{d_{AL}} dz\, p(z) = \Pr(z < d_{AL}) \tag{4}$$

Hence, the dose rate action level is given directly as a quantile of $p(z)$. Similarly, the operational intervention level can be approximated by

$$\Delta E_{IL} = E\left(\frac{\Delta E}{d} d_{OIL} \,\middle|\, d = d_{OIL} \right) \approx E\left(\frac{\Delta E}{d} d_{OIL} \right) \tag{5}$$

leading to the simple relation

$$\frac{1}{d_{OIL}} = E\left(\frac{1}{z} \right). \tag{6}$$

The approximations above require the central range of avertable doses to include the intervention level ΔE_{IL}. Note that the probabilistic approach to calculate the OIL is different from a deterministic calculation where z is taken e.g. as the most likely value; the fluctuations in z will tend to lower the value of the OIL.

For foodstuff restrictions, rather than the avertable dose it is more appropriate to consider radionuclide activity concentrations in the foodstuffs. Interventions is called for when the activity concentrations exceeds a (intervention) level c_{IL} (Bq/kg). We may still apply the definitions (4), (6) for the dose rate AL and OIL, now with the variable z describing the ratio of measured dose rate to the activity concentration in the food item, $z = c_{IL}\, d/c$.

Case study: long-range atmospheric dispersion

Although Eqs. (4) - (6) are fairly general, the method is primarily aimed at emergency situations with long-range atmospheric dispersion of radionuclides. In a case study [3], a countrywide network of gamma monitoring stations detects a radioactive plume, originating from a large nuclear facility. The stations record gamma dose rates and provide online information to the emergency management allowing for rapid implementation of countermeasures.

To calculate the ratio $d/\Delta E$ it is assumed that the relative radionuclide concentrations in the plume depend only on the release characteristics and the transport time, while details of the atmospheric dispersion and deposition are ignored. The accident scenarios are taken as major releases from a RBMK-type NPP. The density of monitoring stations is assumed to be sufficient for one of the stations to measure a dose rate at least at a level, which is 10% of the maximum dose rate within the affected area. The pdf $p(z)$ is readily obtained from the joint probability distribution of the input parameters by Monte Carlo sampling.

A few examples of the pdf are shown in Fig. 2. Since in most cases a single or a few radionuclides is found to dominate the observed external gamma dose rate as well as the avertable dose/activity concentration in foodstuffs, $p(z)$ is well described by lognormal distributions as shown in the figure.

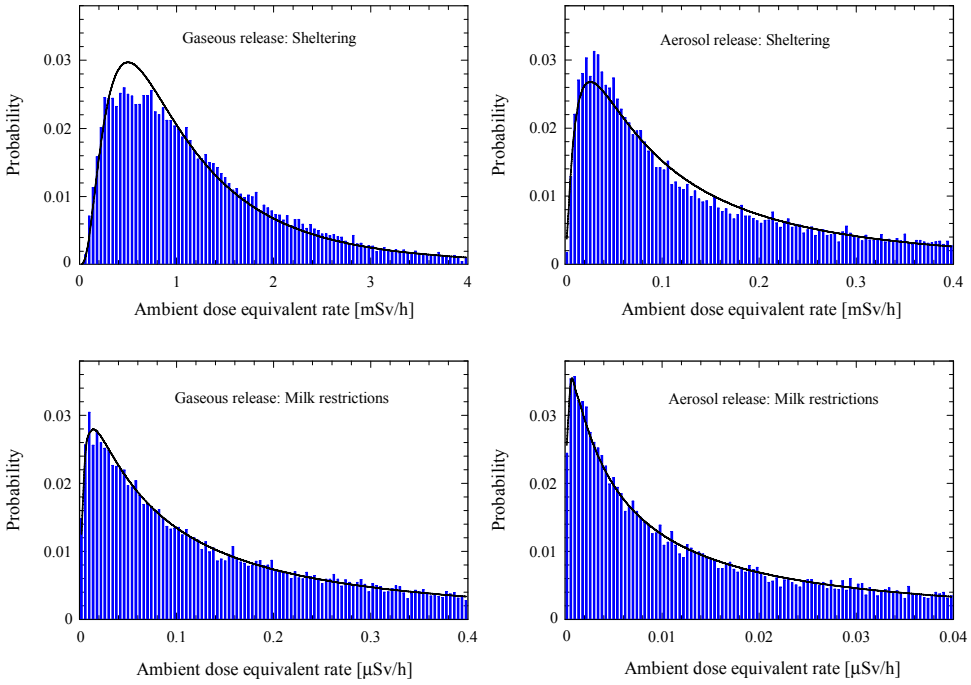

Figure 2. $p(z)$ *for sheltering and milk restrictions. Note the different scales of dose rate.*

Examples of dose rate action levels are shown in Table 1. For sheltering the ALs range from a few tens of microsievert per hour for a large accident to a few hundred in case of a release of only noble gases and organically bound iodine. The radionuclide activity concentrations in foodstuffs vary considerably, depending on the time of year and nature of deposition. For deposition in the growth season, the dose rate action level may be as low as one nSv/h, stemming from the transfer of iodine to milk and leafy vegetables.

The OILs in most cases fall in between the 10% and 20% quantiles of the distribution $p(z)$. Hence, unless the probability r_{AL} is chosen to be considerable smaller than 10%, one may prefer not to distinguish between d_{AL} and d_{OIL} in nuclear emergency planning.

Table 1. Dose rate AL ($r_{AL} = 5\%$) and OIL for two accident release scenarios (μSv h^{-1})

		Gaseous release		Aerosol release	
		d_{AL}	d_{OIL}	d_{AL}	d_{OIL}
Sheltering		260	680	17	59
Foodstuff	beef	65	150	.070	.16
restrictions:	lamb	43	98	.011	.027
	leafy vegetables	.032	.073	.001	.003
	milk	.022	.049	.002	.004

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Measurements in emergency situations aiming at assessment of internal doses

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Abstract: Internal doses can be assessed using results from direct measurements or by using information on activity concentrations in inhaled air and in foodstuffs combined with inhalation and consumption data. Whenever possible direct measurements should be done as soon as possible after an alert to give support for decision making and to reassure the general public. Experiences from accidents where many people have been internally contaminated by radionuclides have demonstrated the need of rapid measurements of large groups of people. The measurements are important both for control of the contamination situation and for later dose assessment.

In the Nordic countries much work has been put down on quality assurance of direct measurements, mostly whole-body counting and on training of dose assessment. This work has been partly funded by NKS. A good network of Nordic specialists has also been created. Nordic emergency preparedness exercises have not included training of rapid measurements of people, thyroid monitoring for I-131 and whole-body for Cs-137, in the early phase of an emergency. There should not be a big gap between planned resources and actual available resources in emergency situations. All planning and instructions are of no use if there are not instruments and people trained to use them available in a sudden accidental situation.

Introduction

Assessment of internal radiation doses can be done using results from direct measurement of people or indirectly by excreta measurements Oliveira et al,IAEA(1994), IAEA(1999), Wallström et al(1999). Especially for iodine estimations can be made using air concentration data in the early phase of an accident. Activity concentration in foodstuffs combined with consumption data can be used later. The aim of measurements is most often to determine the intake of radioactive substances. The internal radiation dose is then assessed using metabolic and dosimetric models.

The external exposure dominates after a nuclear explosion. In a nuclear power plant accident inhalation is the dominating route of exposure at least during the first days after the accident. Ingestion via food and drink becomes important after the radionuclides enter the food-chains. This depends strongly on the time of the year and the growing season.

In cases with high internal contamination the purpose of measurements is to help in deciding if medical treatment or other types of measurement for more exact dosimetry is needed. The measurements are also important for future dose assessment. In situations with prolonged exposure repeated measurements are recommended. Whenever possible direct measurements should be started as soon as possible after an alert to give support for decision making and to reassure the general public.

An alternative method would be to collect urine samples from people in a certain region. The samples could be pooled and sent to another place even far away for measurement. This would give acceptable results for groups in that region but with a certain time delay..

A handbook in Swedish was published as the result of a Nordic NKS project Rahola et al(2002). The handbook contains instructions and descriptions of procedures to handle emergency situations. Supplementary information on detailed instrument specific instructions will further be needed. Mainly iodine and caesium contamination were taken into account in this first version of the handbook. The handbook can be regarded as part of a preparedness plan but it should not be used as such but adopted to local circumstances.

This paper concentrates on direct methods of measurements of people and gives general advice on what to take into account in emergency preparedness planning and on how to do dose assessment.

Methods

After adequate efficiency calibration many simple hand-held instruments can be used for measurement of internal contamination after accidents resulting in environmental contamination. Registration of a few kBq ^{131}I in the thyroid is possible using instruments with NaI-detectors corresponding to an effective dose below 1 mSv. The corresponding data for ^{137}Cs are body burdens of 20-50 kBq. Instruments with GM-detectors need about 5 times higher activities for detection. Lately light weight gammaspectrometers have appeared on the market. Two such instruments are described in another paper presented at this conference Muikku (2005).

To gain most from the thyroid measurements the procedure should be started without delay. An important protective measure is to administer stable iodine in case of exposure to radioactive iodine nuclides. The protective effect is strongly dependent on time of administration. The effect decreases if the tablet is administered too early or too late regarding time of exposure. In an expected fallout situation the iodine tablet should preferably be taken 1-6 hours before exposure. When more than 12 hours has elapsed after intake administration of stable iodine no longer decreases the radiation dose to the thyroid. After the early phase thyroid measurements could be repeated weekly and whole-body measurements of caesium four times a year.

A good review of available software for calculation of internal radiation doses was presented by Ansoborlo et al in 2003. It is necessary to train to use these programs in routine everyday work to be able to do the calculations in a stressing emergency situation. Rapid estimations can naturally be done using dose factors found in ICRP recommendations (1994,1995,1996,1997), IAEA (1996) and in national regulations.

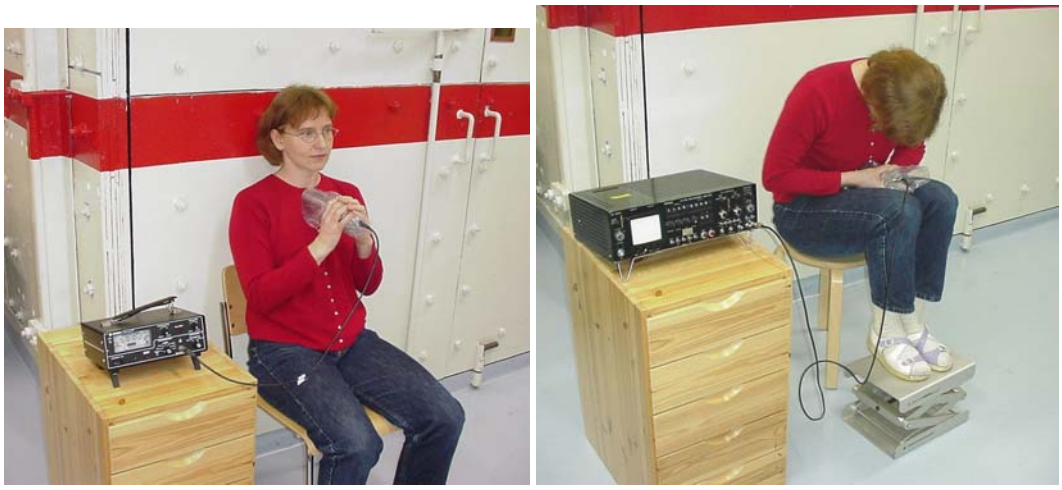

Figure 1 Left: Thyroid measurement with a hand-held instrument based on a NaI(Tl)-detector. Right: A rapid whole-body measurement with similar instrument.

Conclusions

The results obtained with the rapid monitoring methods can be used for different purposes such as:

1. decision of need to administer stable iodine at places to which the radioactive cloud arrives later
2. decision of need to make repeated measurements with higher precision
3. actions in a later phase, especially the protection of certain critical groups
4. later epidemiological studies.

This type of monitoring should be arranged at easily accessible places such as hospitals, health care centres, schools etc. Preferably the place should be in an uncontaminated or slightly contaminated area. From regions with high contamination levels it is better to transport people to be measured to areas better suited for measurement.

To ensure a rapid start of measurement procedures the equipment to be used should be calibrated in advance and also tested regularly. The performance of measurements requires that trained users can rapidly be sent to places decided after assessment of the situation in question. The plans for measurement strategies should be developed in each country separately taking into account national regulations, local circumstances and resources. Nevertheless, in view of the close contacts between people in the Nordic countries, it is very important to have the same approach in all countries. This will increase the public reassurance in case of a nuclear emergency situation. The bilateral agreements among the Nordic countries about assistance in emergency situations also emphasize the importance of cooperation. It is also important to remember that there should not be a big gap between planned resources and actual available resources in emergency situations. All planning and instructions are of no use if there are not instruments and trained people to use them in such demanding situations. In the NKS project IRADES an inventory of available instruments for thyroid monitoring has been done and measurement strategies are developed.

After the accident in Chernobyl an active interest in improving whole-body counting techniques was seen. Intercomparison exercises were started by different organisations in

many countries Rahola et al(1994),Thieme et al(1998). This is most important from a quality control point of view. Quality control aspects are as important when doing rapid measurements as when doing normal whole-body counting. Special instructions should be prepared for different types of measurements in emergency situations and regular training arranged.

In many countries the general public knows that it is possible to do direct measurements on people and will not accept prognoses based only on external radiation and foodstuff measurements. In the future it will be necessary to do also direct measurements on people for reassurance of the public even if such measurements would not be necessary from a strict radiation protection point of view.

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Swedish radiation protection and emergency planning goes east

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Abstract: For more than a decade, the Swedish Radiation Protection Authority (SSI) has participated in the Swedish bilateral assistance co-operation programme with countries of Central and Eastern Europe. The prioritized countries for the SSI support have been Estonia, Latvia and Lithuania, together with the north-western part of Russia. So far, the latter has been supported to a lesser extent, when compared with the Baltic countries. Now that the Baltic Countries have become Members of the EU, this means that the focus of the SSI bilateral programme in the field of radiation protection and radiological emergency planning is successively shifting to Russia. Discussions are going on to enlarge the programme so as to include radiation protection authorities and organisations in Ukraine, Belarus and some of the countries in the Western Balkan.

The SSI assistance programme comprises support for official authorities and industrial support, with the purpose of contributing to the development of radiation protection activities for protecting workers and the public, including the building up of institutions. During the latter part of the 1990s, one task of growing importance focused on activities that were aimed at facilitating the integration of the Baltic Countries into the European Union.

Organizationally, the SSI assistance activities are managed by a special Department for International Development Cooperation (SIUS), separated from the supervisory work of the authority. This report gives some examples of projects in the fields of natural radiation, medical radiology and nuclear power, the latter including the management of radioactive waste and radiological emergency planning.

Baltic Countries

For the Baltic Countries, the break up of the Soviet Union meant that most of the functions provided by the Soviet system of authorities disappeared. Added to this, experts in the field of radiation protection, controlled from Moscow, disappeared from the scene. It became necessary, therefore, to develop new legislation and establish new radiation protection authorities in Estonia, Latvia and Lithuania. In addition to this, there was a need to deal with the various acute technical problems of radiation sources and radioactive waste management, in order to limit the risk of radiation being exposed to the population and polluting the environment. Today, the Baltic Countries have well-established authorities for radiation protection as well as national legislation, in conformity with the EU and their membership in it.

Natural radiation

Co-operation in the fields of natural radiation and radon was initially given priority by the Baltic Countries. The results of this are that the authorities and other organisations have now acquired their own instruments for measuring radon in buildings and the soil, together with the expertise for evaluating and assessing measures that need to be taken in

dwellings and workplaces for reducing any exposure to the population. The projects have included advice to the authorities from Swedish experts on the drawing up of regulations and instructions concerning measures that should be taken to avoid high individual and collective doses from elevated radon concentrations.

Measurements have been carried out, with the aim of locating dwellings, nurseries and schools having a high radon content. Particularly in Estonia, an extensive programme was initiated through the Estonian Geological Survey, for preparing a nation-wide radon chart, with participation and advice from the Swedish Geological Survey and the SSI.

In Latvia, one of the first tasks carried out with Swedish assistance was to investigate the exposure from radon and gamma radiation at sanatoriums that had radon therapy as a speciality, with radon baths and radon-rich drinking water. The investigation led to the radon therapy activities being stopped.

In Salaspils, Latvia, a calibration room has now been established with detectors for radon measurements. This serves as a regional resource for the Baltic Countries.

Radiation protection in medical care

Radiation therapy, X-ray diagnostics and nuclear medicine each have their own special radiation protection problems that need to be dealt with. Experience gathered from certain of the initial projects in the three countries led to a revision of the SIUS programme. This led to focusing activities on training hospital physicists, who, with a few exceptions, were lacking in hospitals.

The training of physicists in medical care was partly carried out in Sweden, where they studied and gained practical experience at the university hospitals in Huddinge, Malmö and Uppsala, among others. The aim of this was that the physicists who received this training will, in turn, assist in the training of new hospital physicists and professional groups involved in medical radiology, in their own respective countries. With regard to longer-term co-operation, the project also aimed at contributing to the creation of networks among experts in radiology and universities.

It was recognised that it was essential to introduce and develop quality systems in medical radiology to support the relevant EU directives being implemented. The aim of these activities was to practically assist in the work of setting up quality systems for medical radiology at a number of local hospitals, so as to act as an example to other hospitals. This project was in common for the Baltic Countries and was managed by hospital physicists from the Central Hospital in Växjö, Sweden.

Nuclear Emergency Planning

From the beginning of the co-operation, the old Soviet era system with regard to emergency planning was in urgent need to being renewed. The aim was to strengthen and broaden the national expertise and ability of the authorities to plan, train and adopt suitable measures for informing and protecting the population if a nuclear accident should occur. The assistance projects have enabled the countries to acquire their own national networks of automatic gamma monitoring stations and some air monitoring equipment. As a result of this, the Baltic countries are now signatories of an agreement concluded among the nine countries around the Baltic Sea and Norway and Island, to routinely provide access to one another’s data from the national networks of monitoring stations.

An important aspect of the projects has been training courses and exercises involving authorities and organisations on central and local levels of the national emergency organisations. As a result of the increased international contacts among the emergency organisations in the Baltic countries and their experts, it was possible, as a finalisation of this part of the programme, to realize a joint exercise in the autumn of 2004. The aims of the exercise were to practise decision-making on the central level and to carry out exchange of information among the three Baltic countries. The exercise was planned, observed and evaluated by a team of experts, including evaluators from Sweden.

Further activities of the cooperation programme are the reviewing of the emergency preparedness plan at the Ignalina Nuclear Power Plant (INPP). These activities included participation by Swedish experts in the role of observers in connection with such exercises. In addition, the SSI provided assistance to the two regional administrative authorities that have responsibilities within the 30 km emergency planning zone around the INPP, Utena in Lithuania and Daugavpils in Latvia. The purpose was to strengthen their emergency preparedness and communication capabilities by support exercises which initiated new contacts among the regional administrations of the two countries.

Viewed from an international perspective, this part of the Swedish assistance programme has contributed towards strengthening a preparedness for international co-operation in the Baltic Sea Region, with improved capabilities for early warning, the interchange of information and assistance, if needed, in the case of a nuclear accident.

Lithuania

Over the years, an extensive part of the Swedish assistance programme has been directed at the INPP, a site with two units of the RBMK type. One of the reactors was taken out of operation at the end of 2004. This will be followed by the second reactor being shut down in 2009. The SIUS projects were designed to improve the radiation protection of workers by improved staff training courses on radiation protection and the use of modern techniques for individual dose measurements. With support from experts from the SSI, a review of the environmental monitoring programme of the INPP was done, with the result that the programme was able to be approved of by the authorities. Equipment was procured for enabling alpha emitting radionuclide samples to be prepared and measured at the site. A project was run which included the training of personnel from INPP at the laboratory of Risö in Denmark. These two activities were jointly financed, in co-operation with the Danish Emergency Board

Comprehensive international support is being given and planned in the areas of the management of radioactive waste and to the decommissioning operations at the INPP, with Sweden being one of the contributors. During the years, the SKI, the SSI and Swedish industry organisations have contributed their own experience with guiding Lithuania in developing a national strategy and methods for the management, storage and disposal of the different kinds of radioactive waste and spent fuel.

In a consortium with the Finnish Safety and Radiation Protection Authority (STUK), the SSI had the project management role for a Twinning project financed by the Commission of the EU Phare programme. The aim was to strengthen the operations at the Lithuanian Radiation Protection Authority (RPC). The Twinning project had a budget of EUR 750,000 and the activities were in progress for a period of 24 months, concluding in July, 2004. During the project, a total of about 30 experts from the SSI and STUK were on

short-time duty at the RPC, with the addition of a radiation protection expert from the STUK being stationed at the RPC in Vilnius in the capacity of an advisor.

Estonia

In Paldiski, about 40 km from Tallinn, the Soviet navy used a plant for training the crews of their submarines. The plant was equipped with two nuclear power reactors (with a thermal effect of 70 MW and 90MW, the criticality ended in December 1989) installed in two full-scale submarine simulators located in a large hangar. When Estonia took over the operation of the plant, several buildings contained solid and liquid radioactive waste and were in such a condition that they were needed to be decommissioned. The comprehensive handling of the waste and the decontamination work has been carried out with the aid of international support actions and Swedish bilateral assistance through the SKB, Studsvik and the SSI.

Russia

The orientation of the co-operation of the SSI with Russia is directed at the federal authorities in the field of radiation protection, such as Rostekhnadzor and the Federal Medical Biological Agency and their respective technical support organisations, the Institute of Biophysics in Moscow and the Institute of Radiation Hygiene in St Petersburg. Examples of the activities include the development of methods, training and tools for carrying out analyses of the environmental impact of nuclear facilities used for the storage of radioactive waste. The projects are aimed at supporting the authorities, by using updated regulations when prescribing the radiation safety requirements in connection with evaluating applications for licences, with regard to different reconstruction work at radioactive waste facilities.

When it comes to industry co-operation, there are projects directed at two nuclear power sites, Sosnovy Bor (LNPP) outside St Petersburg and Polyarnie Zori (KNPP) on the Kola Penninsula. A prioritised action in this program was to support the replacement of the old individual dose measurement systems with modern TL-dosimeter equipment. For the exchange of experience, radiation protection experts from the two plants have visited NPPs in Sweden. The visits included a period when the reactors are in normal operation and when an up-coming revision is being planned, as well as an outage period when most of the occupational exposures take place. Another project in progress with the LNPP is on issues connected with monitoring and measuring releases of radioactive substances from the plant.

Two other projects directed at the industry concern the reconstruction of solid radioactive waste storage sites at Leningrad´s Specialied Kombinat “RADON” and the Murmansk Special Plant “RADON”. The radioactive waste at these sites comes predominantly from operations outside the nuclear field. The waste storages are in poor condition and there is a need for action to avoid the environment and the groundwater being polluted by radioactive substances.

Information issues in Russia are also considered through cooperation with the Rosatom (earlier Minatom) Information Centre (SREC) in St Petersburg. The programme includes problems in communication with and information to the public, regarding issues related to the nuclear power industry and risk communications. Specifically, some of the activities have covered the preparedness for handling public information matters in case of a nuclear accident.

Source term estimation from off-site radiation monitoring data

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Abstract. A Kalman filter method intended for real-time estimation of the source strength and the short-range atmospheric dispersion of radionuclides by means of automatic off-site gamma radiation measurements has recently been developed. The method has been tested using experimental radiation monitoring data and proves rather accurately to recover the significant dispersion parameters including the source term. In case of a nuclear accident, the source term estimates may be used as input for both long- and short-range atmospheric dispersion model calculations and will improve assessment of the radiological consequences of the accident.

The Kalman filter

Recently, a Kalman filter method was proposed as a tool for real-time estimation of the radionuclide source term and the significant atmospheric dispersion parameters, based on local off-site radiation monitoring data, cf. Drews *et al*. (2004). The Kalman filter is a predictor-corrector algorithm, where the model parameters are recursively updated by current measurement data, weighting the difference between the model predictions and the observations by an optimal gain factor. The filtering procedure is computationally efficient, allowing updated estimates to be calculated as soon as new measurements become available.

The Kalman filter method is based on a multivariate nonlinear stochastic state space model formulation. In its general form, a system equation describes the dynamics of the atmospheric dispersion model parameters, i.e. the state variables, $X = (x_1, \ldots, x_n)$,

$$X_t = f(X_{t-1}, \ldots) + w_t, \tag{1}$$

while a static measurement equation describes the coupling to the physical observables, e.g. dose rate measurements,

$$Y_t = h(X_t, \ldots) + v_t. \tag{2}$$

The set of observables Y_t comprise radiation and meteorological measurement data obtained prior to and during the accident. The system and measurement errors, w_t and v_t, are uncorrelated, stochastic Gaussian white noise processes.

The Kalman filter method provides a best estimate of the model parameters X_t. The embedded parameters of Eqs. (1), (2), i.e. the system and measurement noise covariance, adjustable parameters of the atmospheric dispersion model, and the initial values of the state variables, $X_{t=0}$, can be determined through maximum likelihood estimation.

Source term estimation

The proposed method has been evaluated numerically using both simulated data as well as radiation monitoring data from an ^{41}Ar atmospheric dispersion experiment, cf. Drews *et al*. (2005). The dispersion of the radionuclides is calculated within the Gaussian plume model, applicable up to a few kilometres from the release point. To keep calculations computationally simple, the dynamics of the state variables, i.e. the radionuclide release rate divided by the wind speed ($\dot{Q}/u$), the plume height (h), the main plume advection direction (θ), and the difference between the observed wind direction and the plume advection direction (θ'), are modelled as an autoregressive process, cf. Eq. (1),

$$X_t = AX_{t-1} + w_t \,. \tag{3}$$

Remaining plume dispersion parameters are prescribed externally. The measurement data, Y_t, in the experiment consists of simultaneous dose rate measurements from four ground-based detectors and the wind direction (Rojas-Palma *et al*., 2004).

In Fig. 1, the Kalman filter results (dashed curves) are compared to experimental observations of the model parameters (solid curves). The numerical results demonstrate that Kalman filtering provides accurate estimates of radionuclide release rate and the significant dispersion model parameters. The standard deviations on the parameter estimates from the Kalman filter is shown in the right column of the figure.

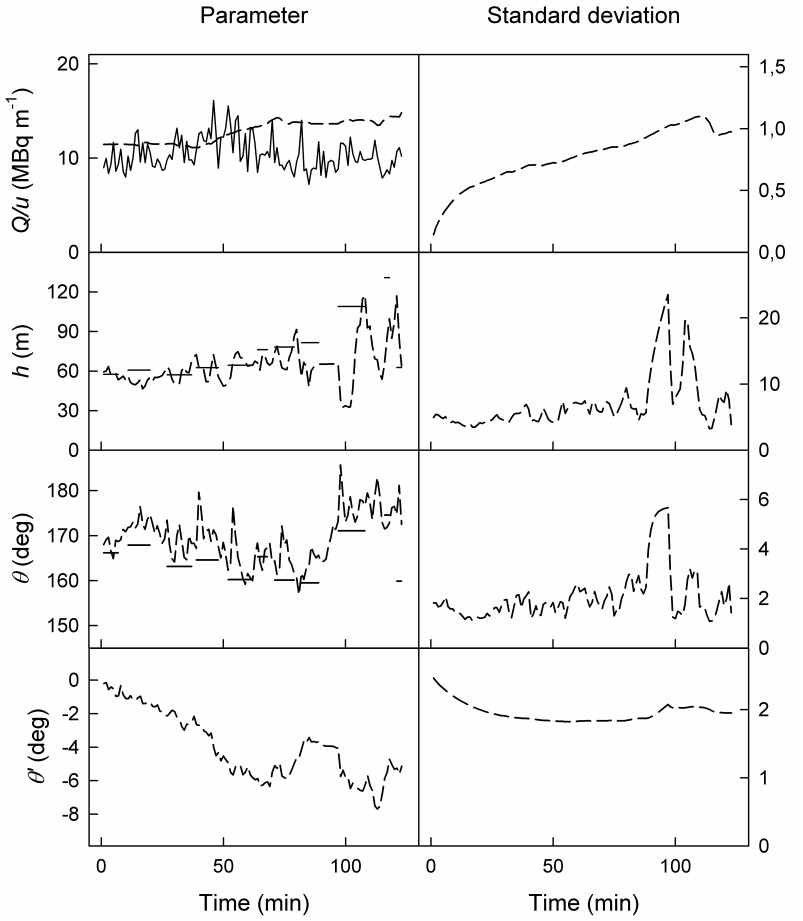

Figure 1. The Kalman filtered state variables and associated standard deviations.

In Fig. 2, the source term estimated from the Kalman filter is compared to direct observations of the released activity divided by the wind speed ($\dot{Q}/u$). In the four time series shown, different experimental setups were employed. For the analysis shown in Fig. 1, the radiation measurements span the plume in the crosswind direction. Even for less optimal detector placement, the Kalman filter provides consistent parameter estimates of the source strength and the plume model parameters, however associated with larger prediction errors. In general, the method is found to be robust and capable of handling outliers as well as missing observations.

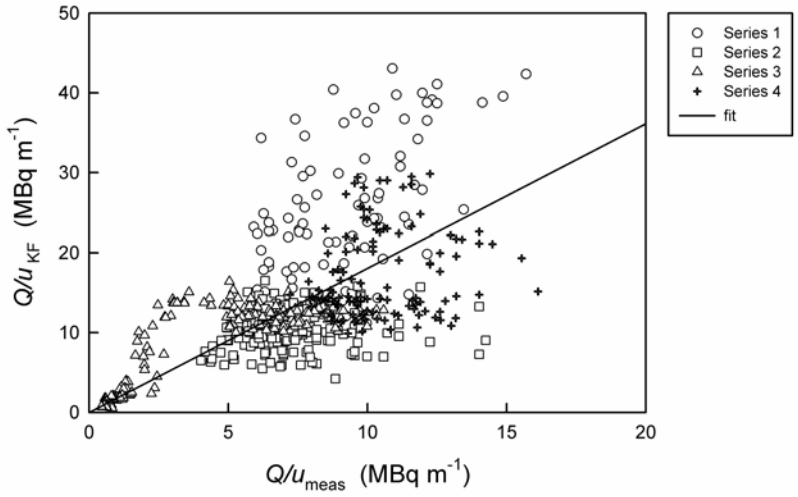

Figure 2. Measured versus estimated source term for four time series of experimental data.

Discussion

The Kalman filter method potentially could be an effective operational tool for real-time estimation of the source strength and short-range dispersion model parameters, based on local off-site radiation monitoring data. The estimated dispersion model parameters, in particular the source strength, may be used as input for long- and short-range atmospheric dispersion models, making an assessment possible of the radiological situation in the emergency phase of a nuclear accident.

A main difficulty lies in determining the embedded state space model parameters. The parameters must be determined either externally, e.g. using the expert judgments, and/or estimated from measurements. In the analysis above, the parameters have been estimated by the maximum likelihood method based on the entire data set. In an on-line automatic decision support system, however, it is essential that calculations be carried out recursively, such that model estimates can be updated at each time step. For fast computing one may rely on pre-defined parameters, assuming that these are sufficiently close to being optimal. For internal estimation, one may use small subsets of data to determine the parameters through maximum likelihood estimation. During “background” source term estimation or for a prolonged release, it may be necessary periodically to re-estimate the embedded parameters because of changes in the meteorological conditions or in the accident characteristics. Only the most recent part of the time series should be used for estimating the parameters. Alternatively, one could employ an empirical Bayesian approach

to incorporate prior information about the embedded parameters directly into the estimation scheme, e.g. using the *maximum a posteriori* method. This will reduce the computational load compared to maximum likelihood estimation. In addition, the numerical optimization may become faster and more robust.

A particularly attractive aspect of using the stochastic state space model framework is that such models are very flexible, allowing for more complex dispersion modeling schemes, while explicitly taking into account the inherent model and measurement uncertainty. As opposed to purely deterministic models, parameter estimates in stochastic models tend to have less bias. The stochastic state space models combine *a priori* knowledge about a physical system with the use of statistical methods for parameter estimation. The Kalman filter provides a direct estimate of the parameter error covariance, which amounts to an assessment of the quality of the predictions. The coupling of the dynamics of the states to physical observables generally ensures that parameter estimates are bounded.

To account for realistic accident scenarios, the atmospheric dispersion model, viz. the simple Gaussian plume model, should be amended to include models for dry and wet deposition. Also, the gamma radiation (dose rate) model should allow for the simultaneous release and decay of many different radionuclides. While such extensions are easily accommodated in the state space model framework, the detailed treatment depends on the measurements available. If only gamma dose rates or gross count rates are measured, the radionuclide composition of the release becomes unobservable. Still, the Kalman filter method can be used to assess the total activity released, if the radionuclide makeup of the plume is assumed known.

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Portable thyroid monitors for detection of ^{131}I in emergency situations

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Abstract: The need for assessing internal radiation doses in emergency situations is evident. Studies done in Ukraine, Belarussia and the southern parts of Russia after the Chernobyl accident have shown an unexpected increase of thyroid cancers in children and young people. Direct measurements give both individual results and results for groups. In the past two years, Radiation and nuclear safety authority - STUK has obtained 35 monitors for thyroid measurements in the field conditions. The monitors work as spectrometers, which makes it possible to do real time spectrum analysis in the field.

Introduction

The aim of this work is to improve the preparedness for thyroid monitoring in emergency situations. The need for assessing internal radiation doses in emergency situations have been demonstrated after accidents in Brasil, Ukraine and other countries. Studies done in Ukraine, Belarussia and the southern parts of Russia after the Chernobyl accident have shown an unexpected increase of thyroid cancers in children and young people. Direct measurements, which should be done as soon as possible after an alert to give support for decision making, give both individual results and results for groups.

Results and discussion

In the recent years, Radiation and nuclear safety authority - STUK has obtained 25 Atomtex RKG-AT1320 monitors and ten Canberra InSpector 1000 monitors for thyroid measurements in the field conditions. The AT1320 thyroid monitor consists of a detector with 1”x1” NaI(Tl) crystal, control unit and lead collimator (Figure 1, left). The construction of the InSpector 1000 thyroid monitor is similar apart from the size of the NaI(Tl) crystals which is 2”x2” (Figure 1, right). The energy resolutions of the crystals are better than 8 %. The monitors work as spectrometers making it possible to do real time spectrum analysis in the field. The collected spectrum is shown on the display of the control unit. It is possible to store spectra in the detection unit, from where they can be later on transferred to a PC. The energy stabilization is done before starting measurements using a ^{137}Cs source. Likewise the background control and sensitivity check (also with ^{137}Cs source) are carried out before measurements.

Figure 1. Left: RKG-AT1320 thyroid monitor: NaI(Tl) detector, control unit, charger, ^{137}Cs control source and cables. Right: InSpector 1000 thyroid monitor: In addition to spectrum acquisition mode the Inspector monitor can be used in dose rate mode or in nuclide identification mode.

The aim of thyroid monitoring is to determine the ^{131}I activity in the thyroid with minimal interference from activity in the rest of the body. This is achieved by placing a shielded or a collimated detector near the neck at the position of the thyroid. However, both RKG-AT1320 and InSpector 1000 thyroid monitors are planned to be movable and easy to carry. During measurement the RKG-AT1320 detection unit planned to be held in hand, although it can also be placed to a stand. Thus, the weight of the detection unit can not be more than a couple of kilograms. The collimator used in the present RKG-AT1320 measurement set-up is made of 5 mm thick lead and it weights about 1 kg (Figure 2, left). Assessment of the contribution to the detector response from radioiodine in blood and in surrounding measurement area can be made by placing the detector over a different area in of the body (e.g. thigh) [1]. For thyroid measurements, InSpector 1000 monitor is positioned into a table-top lead shield of 12.5 kg weight (Figure 2, right).

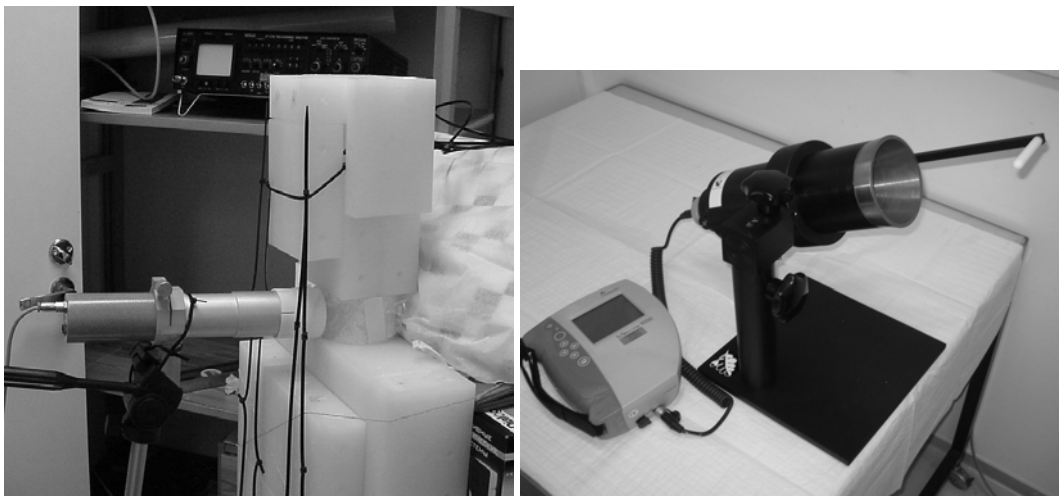

Figure 2. Left: RKG-AT1320 thyroid monitor in a 5 mm lead collimator and an adult phantom in measurement distance of 7 cm. Right: InSpector 1000 thyroid monitor in a table-top lead shield.

The detection efficiency depends greatly on the distance between the detector and thyroid. To get good statistics in a short time the detector should be placed as close to the neck as possible. The exact position of the thyroid must be known in order to get right results in the close geometry. If the detection distance is longer, is not so crucial to place the detector exactly on the thyroid. Two measurement distances were used in the calibration of the RKG-AT1320 thyroid monitors, 7 cm and 20 cm. Three calibrations were done for both distances: adult, teenager (14 years old) and child (6 years old). The St. Petersburg thyroid phantom and whole body phantom with ^{40}K rods were used in the calibration. Two capsules filled with ^{131}I solution were placed in the thyroid phantom. The minimum detectable activities (MDAs) when using the measurement time of 100 s in normal conditions are about 2000 Bq and 330 Bq for 20 cm and 7 cm detection distances, respectively. If longer time of 600s is chosen, the MDAs are significantly lower: 760Bq and 120 Bq for 20 cm and 7 cm distances, respectively. The set-up with 7 cm detection distance is about six times more efficient than that with 20 cm distance.

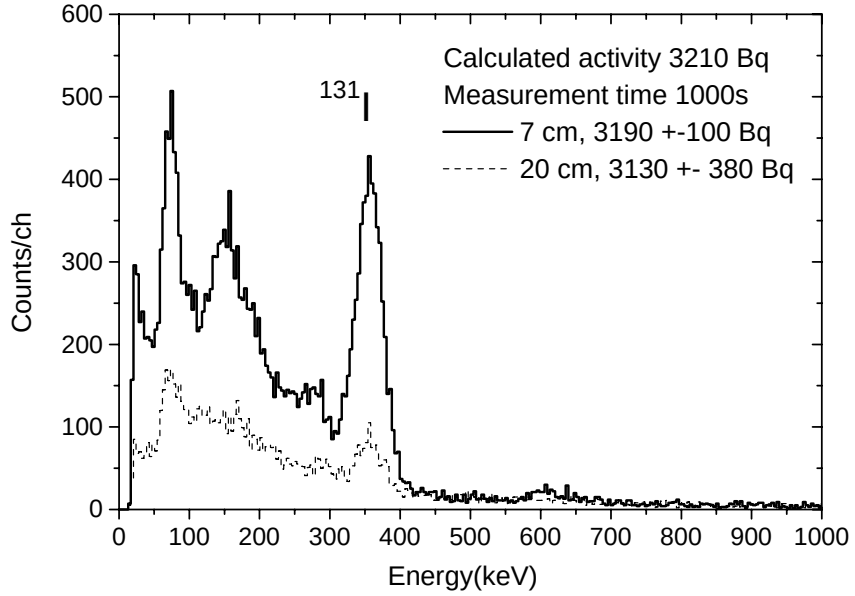

Figure 4. Gamma-ray energy spectra collected from a ^{131}I thyroid phantom (distances 7cm and 20 cm).

The use of a lead collimator - even though the thickness of the collimator is only a few millimetres - reduces significantly the low-energy background resulting from scattered γ-rays. This is illustrated in Figure 5, where a γ-ray energy spectra collected from a ^{131}I thyroid phantom with and without the 5 mm lead collimator is shown. A ^{137}Cs source has been placed in the vicinity of the thyroid monitor to demonstrate the background radiation from surroundings in a fallout situation. These tests have also been carried out using RKG-AT1320 thyroid monitor. InSpector 1000 monitors will be calibrated and similar tests will be done during summer 2005.

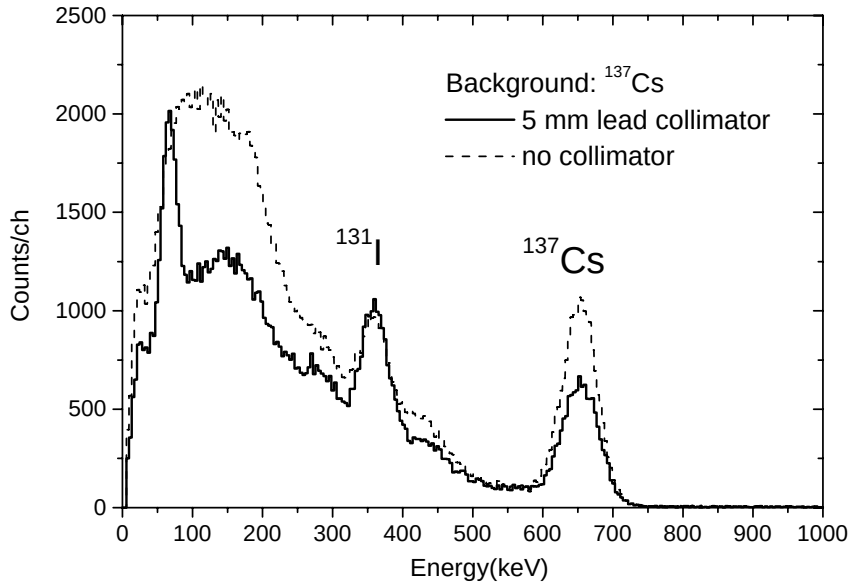

Figure 5. Gamma-ray energy spectra collected from a ^{131}I thyroid phantom with and without the 5 mm lead collimator.

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Nordic decision support handbook for contaminated inhabited areas

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Abstract: Under the framework of NKS, a handbook is being produced, with the objective of assisting Nordic decision-makers in addressing the consequences in inhabited areas of severe nuclear accidents. It will contain descriptions of a compilation of countermeasures, that are deemed to be particularly relevant for reduction of long-term doses from contamination in Nordic living areas. It will also discuss optimisation of countermeasure strategies, including easily applicable methodologies employing state-of-the-art data for estimation of doses.

Introduction

Before the Chernobyl accident, very little thought had been given to the implications of radioactive contamination of cities, towns and villages, although this is where most people spend most of their time. The reason for that is probably the erroneous assumption that the consequences of conceivable nuclear incidents would be relatively short-term and restricted to rural areas. However, the Chernobyl accident demonstrated that large nuclear accidents can lead to significant long-term contamination problems in vast inhabited areas, including cities. It has in fact been estimated after the first decade following the Chernobyl accident that there was an equal contribution in the strongly contaminated Bryansk region of Russia from internal and external doses (Balonov, 1997). It is the objective of the NKS/URBHAND project to create a handbook designed to assist Nordic decision-makers in the remediation of contaminated inhabited areas in the event of a severe nuclear emergency. Although the focus is generally on nuclear power plant accidents, this handbook, contrary to previous works for assisting in remediation (e.g., Howard et al., 2004; Andersson et al., 2003), to some extent also considers the implications of malicious dispersion devices. In recent years there has been an increasing awareness of the threat posed by so-called ‘dirty bomb’ dispersion devices. It seems evident that a terrorist would obtain maximum impact on society by detonating such a device in an urban area. 'Dirty bombs' may lead to very inhomogeneous contamination patterns, and as a range of radionuclides might be applied for the purpose, these types of emergencies can be quite different from those resulting from nuclear power plant accidents.

Methods/ results

The handbook will contain different parts of information that is deemed to be valuable in a decision-making process. This information is foreseen to comprise the following key elements:

Organisational structure of Nordic emergency management: In relation to the specific task of Nordic emergency management for the remediation of contaminated inhabited areas in the Nordic countries, it is important to have a clear definition and overview of the tasks that the various key actor organisations would carry out. Responsibilities may conveniently be distinguished between:

- Coordination and supervision
- Technical support and advice, including field measurements, monitoring and evaluation of countermeasures
- Decision-making and formulation of countermeasure strategies
- Implementation of clean-up actions and interaction with affected population groups and organisations
- Information and dialogue with, e.g., the public and media

Lists of responsibilities on national scales in the Nordic countries will reveal differences and similarities between the various forms of organisation across borders. This is valuable for mutual understanding and to foster harmonisation and encourage reconsideration.

Strategies for measurement and mapping: Monitoring in connection with a nuclear incident affecting inhabited areas can have a number of different purposes. It is essential that planners are aware of the needs for monitoring in different situations, so that it can be ensured that the preparedness system is operative. Potentially important elements include:

- Screening of areas of large dimensions. This can give valuable information on the large-scale contamination pattern of gamma emitters. On this basis, it is possible to focus resources where they are most needed in the early phase.
- Dose rate measurements. Such measurements can give a rough idea of the local extent of the emergency, but generally give no information on how the sources contributing to this dose rate are distributed in the area.
- Measurements to determine contamination levels on specific types of surface in the environment. These can imply a combination of sampling and collimated in situ measurements, and will give detailed information on the contamination situation and its future implications.
- Monitoring to assist in optimising countermeasures and show that they have had the desired effect. These have been demonstrated to have high importance (Fogh et al., 1999).
- Personal monitoring and monitoring to provide control/assurance.

Calculation of doses: In the event of a contaminating incident it is important to be able to rapidly estimate the magnitude of the doses people may receive. This is valuable in judging the severity of the situation and its future implications. Estimates of dose contributions received after the emergency phase can form a platform for decision making, ensuring the identification of optimal countermeasure strategies. The methodology given in the handbook is aimed at enabling the decision-maker to rapidly obtain an overview of the various dose contributions in an affected area. This easily

applicable methodology can be further supported by the use of more detailed consequence assessment systems such as the European ERMIN model, which is currently being developed for implementation in the RODOS and ARGOS systems. Dose contributions that are considered in the handbook include:

- External gamma doses from contamination on outdoor surfaces in the inhabited environment.
- External gamma doses from contamination on indoor surfaces.
- External beta doses from contamination on outdoor as well as indoor surfaces.
- Gamma and beta doses from contamination deposited on humans.
- Doses from inhalation of contaminants resuspended after the contaminating incident has occurred.
- Doses from ingestion of contaminated food grown in inhabited areas. E.g., considering routine soil management techniques applied for small kitchen garden lots are very different from those applied in large areas of industrial agriculture.

Description of countermeasures and strategy formation: The handbook will comprise descriptions of a total of 17 countermeasures, which are believed to be the most suitable options for reduction of long-term external doses in Nordic inhabited areas, following a contaminating incident. Contrary to previous countermeasure databases (e.g., EC-STRATEGY; Andersson et al., 2003), the handbook to some extent also considers the implications of malicious dispersion devices. The selected countermeasures cover the 6 different important types of surfaces in an inhabited environment: open (grassed) soil areas, paved areas (streets), house walls, house roofs, vegetation (trees, shrubs, bushes), and indoor surfaces. Systematic and holistic countermeasure descriptions are valuable to speed up the decision-making process and at the same time ensure that potentially important issues are not overlooked in the process of optimisation. The descriptions are aimed at providing an overview of methods and factors affecting their applicability in a standardised format facilitating intercomparison. Such descriptions allow preparedness planners to swiftly assess which countermeasures would be most suitable/acceptable given the specific conditions in question, e.g., with respect to topography, building tradition and soil type. The descriptions also clarify which requirements each of the countermeasures have, so that it can be ensured that the local practical preparedness is operational. A series of non-technical potential problems of countermeasure implementation are discussed. This may for instance be useful in defining public interaction plans. Also, as stressed by the ICRP, non-radiological costs and benefits are essential to include in optimised preparedness strategies. In addition, a number of countermeasures specifically targeted at reducing doses from ingestion of contaminated kitchen garden products are discussed.

Waste management: In choosing between different waste management options, it is important to identify methods that are both safe and inexpensive. For instance, certain minimum safety requirements would need to be fulfilled in the design of a final waste repository. However, disposal in expensive deep ground repositories, as may be required for some highly contaminated wastes originating from practises, is not likely to be realistic in connection with waste from cleaning of contaminated inhabited areas. A number of processes for waste management prior to final storage are discussed:

- Loading and transportation of wastes
- Storage of waste if advantageous with respect to short-lived radionuclides
- Filtration of solid particles out of liquid waste
- Treatment for contaminants in liquid solution
- Reduction of volume of organic waste
- Stabilisation of solid waste preventing movement of contaminants

Also recommendations for construction of waste repositories are given, with check-lists of important things to consider.

Conclusions

A handbook for decision-support in contaminated inhabited Nordic areas is under development. An outline is given of some of the key features of this handbook, which is aimed at improving the decision-platform for responsible organisations in the Nordic countries. The organisational structure in emergency management is outlined, and recommendations are given concerning measurement strategies that would be useful in addressing a contamination problem. Also, simple methodologies for calculation of potentially important dose contributions are given, which can be used to analyse solutions to the problem. Descriptions are given in a standardised format of the countermeasures that would be deemed most suitable for Nordic inhabited areas, including kitchen gardens. Methodologies for processing and disposal of the wastes generated by such countermeasures are described. It is envisaged that the prototype handbook, which is due by the end of 2005, will be further customised through interactions with national end-user fora in Denmark, Finland, Sweden and Norway. This will involve testing in an application exercise.

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Radiation protection inspector on duty

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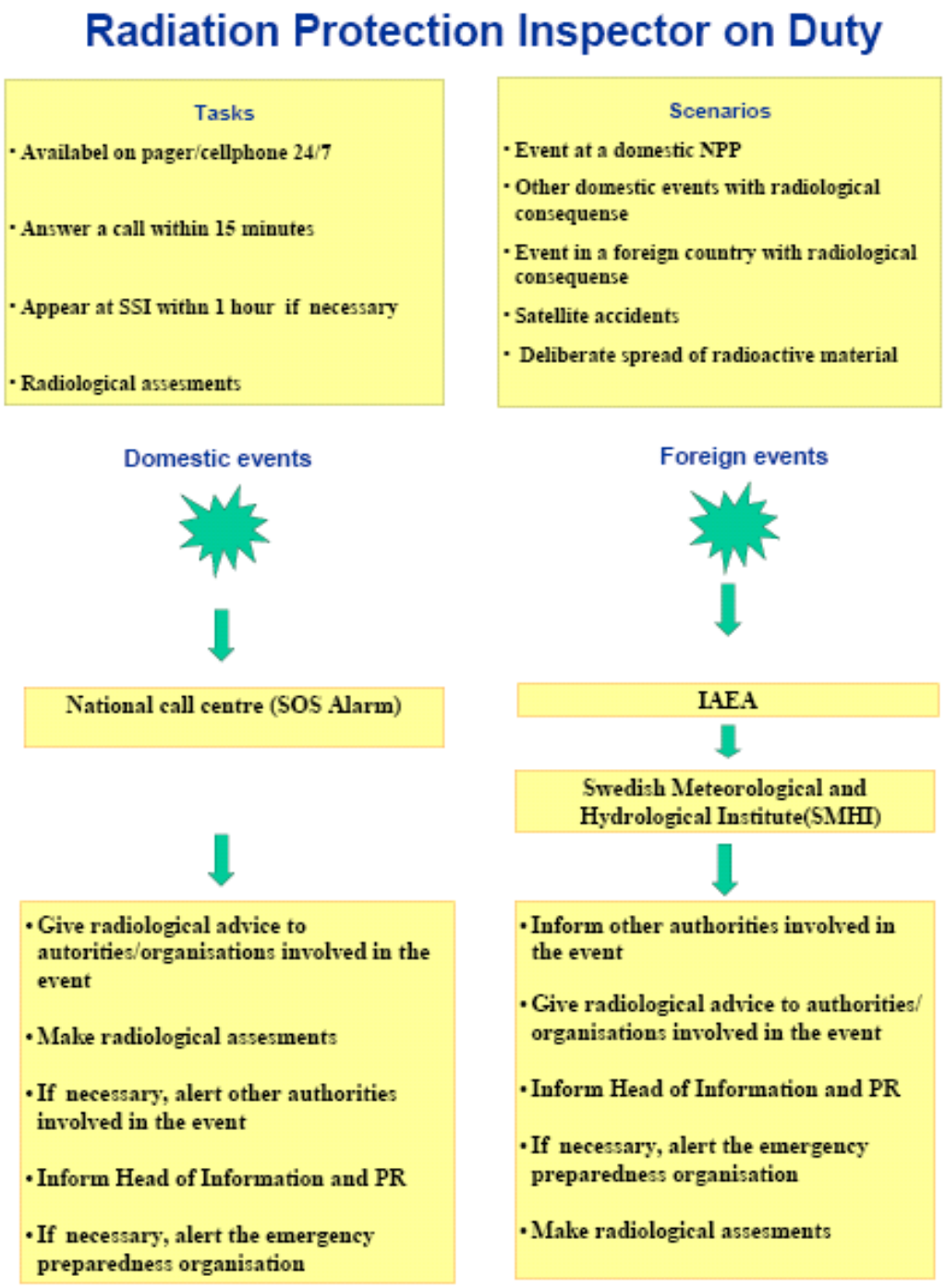

Installation of radiation detection portals at the Norwegian border crossing station at Storskog

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Abstract: During 2001, work was initiated to improve Norwegian emergency preparedness with respect to acts of terrorism. This effort resulted in, among other things, the installation of radiation detection portals at the Storskog border crossing station on the Norwegian-Russian border. The Norwegian Radiation Protection Authority (NRPA) has been involved in the process of establishing the current monitoring system, and the NRPA Emergency Preparedness unit at Svanhovd provides the station with scientific expertise and in-situ assistance when needed.

Introduction

The dissolution of the Soviet Union and changes in the nature of the threat posed by international terrorism have made the theft and illicit trafficking of both fissile and non-fissile radioactive material increasingly important issues. Norway has for some time been active in non-proliferation work and established in 1995 a national board to prevent illicit trafficking of such material, with representatives from the Norwegian Defense, the police, the Norwegian Customs and Excise and the Norwegian Radiation Protection Authority (NRPA). However, the Norwegian concern at the time was primarily on the possibility of proliferation of fissile material to aspiring nuclear weapon states.

With the recent changes in the international security environment, awareness has been heightened towards the possibility of increased risk of illicit trafficking of both fissile and non-fissile radioactive material to the West, including across the Norwegian-Russian border. Hence, when an initiative was made to improve the Norwegian emergency preparedness with respect to acts of terrorism subsequent to the attack on the World Trade Center in 2001, this included emphasis on the possibility of malicious use or illicit trafficking of radioactive material as well. The newly installed radiation detection portals at Storskog are a result of this initiative.

The station at Storskog is the only border crossing station on the Norwegian-Russian border; it also is the only Norwegian border crossing station on the outer border of the EU/EEA area. The station is situated 7 km east of Kirkenes and is attended by the police, the Norwegian Customs and Excise and representatives from the Norwegian Border Commissioner. The installed portals are owned and operated by the Norwegian Customs and Excise.

The radiation detection monitoring system was considered operational from early 2004, following a set of independent acceptance tests by both the contractor and the NRPA.

The monitoring system

The proposal regarding improvement of the national Norwegian border control regarding radioactive material was considered by the Norwegian Parliament in June 2002, and approximately € 375,000 was allocated to the improvement. It was decided to use these funds on a high quality system at the Storskog border crossing station. The system was expected to meet a set of requirements, including IAEA recommendations as described in IAEA (2002). Furthermore, for practical reasons it was decided that the monitoring system should include in total three radiation detection portals, one for each of the customs lines. These portals monitor all traffic arriving to Norway at the station.

The installed portals at Storskog are of an "Exploranium" type, with two detection modules in each portal. The detection modules are capable of detecting both gamma and neutron emissions. The customs personnel are able to continually monitor the nature and level of radiation from their control room. If the radiation levels exceed a set limit, the system alerts the customs personnel through both audible and visual alarms and displays the approximate radiation level and location of the radiation source within the vehicle.

In addition to the radiation detection portals, the monitoring system also includes a set of portable survey instruments of an "Identifier" type. These instruments enable the customs personnel to locate, identify and verify suspected radioactive material suggested by the portals, as well as to inspect vehicles manually. These instruments detect and identify both gamma and neutron emitting radionuclides.

Figure 1. One of the detector modules without its cover; the modules contain a PVC scintillator for gamma ray measurements and three helium tubes for neutron detection (Photo: NRPA).

The NRPA involvement

The NRPA has worked in close collaboration with both the police and the Norwegian Customs and Excise during the process of specifying the requirements for the radiation detection system, selecting it and installing the detection portals. The NRPA has arranged

several training courses for the local customs personnel and has prepared a related manual, NRPA (2005), providing background information and describing proper actions to be taken upon discovery or suspicion of radioactive material.

Furthermore, the NRPA Emergency Preparedness unit at Svanhovd is located near the Storskog border crossing station, and serves as a second line service for the station. The unit has regular tests of the equipment, and provides the station with both scientific expertise and in-situ assistance when needed. The unit is continually informed of the performance of the system and of any alarms, and is at any time able to electronically connect to the system in order to assess and comment on any particular measurement.

Experience with the monitoring system

Since the radiation detection portals were initially used in November 2003, the system has produced alarms on several occasions. Most of these alarms have been 'false' alarms, in the sense that no actual radioactive material was present. This type of alarm occurs during large variations in the background radiation level, commonly due to adaptation by the system to different shielding attributes in vehicles. On occasion, the system has produced 'innocent' alarms, where legitimate transportation of non-fissile material with coincidentally elevated radiation levels was detected by the monitoring system. So far these alarms have exclusively been caused by transportation of Russian reindeer meat contaminated with caesium-137; however, transportation of scrap metal, wood, rock, soil or ore etc. also could cause similar alarms.

Until now, there have not been any 'real' alarms, where radioactive material of concern has necessitated intervention by the authorities.

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Air monitoring stations in Norway administered by the NRPA

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Abstract: The NRPA currently administers five air filter stations across Norway in order to monitor the atmospheric environment. These stations have been established through the last decades due to an increasing attention to nuclear emergency preparedness and the need for surveillance of radionuclide concentrations in the atmospheric environment. They have become an important tool to trace potential releases of radionuclides but are also used to study radionuclide behavior in the atmosphere.

Introduction

The Norwegian Radiation Protection Authority (NRPA) administers five air monitoring stations. These stations are an important constituent in the authority's surveillance of radionuclide concentrations in the atmospheric environment.

The stations are situated at the following places:

	Place	*Position (approx.)*	*Running since*
1.	Østerås (NRPA's head office)	59° 55' N, 10° 33' E	1980
2.	Stavanger (west coast)	58° 52' N, 05° 37' E	2002
3.	Skibotn (outside of Tromsø)	69° 22' N, 20° 17' E	1975
4.	Viksjøfjell (close to Russian border)	69° 36' N, 30° 44' E	1995
5.	Svanhovd (NRPA's preparedness unit)	69° 28' N, 30° 03' E	1993

All stations have the same principle for air sampling. However, some stations are more efficient and more reliable than others due to differences in manufacture and age of the equipment. Common for all stations is that air is pumped through a glass fiber filter with high density, in order to collect aerosols with associated radioactive particles from the air. The mesh size of the filters is 2 μm. The filters themselves are changed once a week and are sent to NRPA for analysis.

The air monitoring stations at Østerås and Svanhovd (close to NRPA's laboratories) are also equipped with a preserved carbon filter for measurement of radioactive iodine gas in the air - this is important from an emergency preparedness point of view.

Four of the five stations are also equipped with a Geiger-Müller tube which is placed directly over the filter. This is a real-time surveillance of the dose rate close to the filter

(RADOS-system). If a predefined limit is reached, this will cause an alarm which will trigger a transmission of an SMS or an email to a predefined address.

Figure 1: Two airfilter stations are placed in the southern part of Norway - three in north.

Preparation and analysis

The airflow capacity for the stations is approx. 900 m^3 / hour, resulting in 120 - 150,000 m^3 of air weekly being sampled at each station. The filters (50cm x 60cm) are prepared in the laboratory by folding the filter and squeezing it a manual hydraulic press. The filter are then transferred into a geometry and counted for 48 hours using a high purity germanium-detector.

Results and discussion

In addition to a number of naturally occuring radionuclides, anthropogenic caesium and occationally iodine is found at detectable levels.

Caesium-137 is frequently detected at all stations, with a noticeable higher concentration in the southern part of Norway compared with the northern part. For the three stations in the northern part the concentration are generally in the range 0.2 - 0.5 $\mu Bq/m^3$. The concentration in the south are commonly 3 - 4 times higher. This is assumedly due to the generally higher concentration of resuspended airborne caesium-137 in the southern part of Norway which dates from the Chernobyl accident.

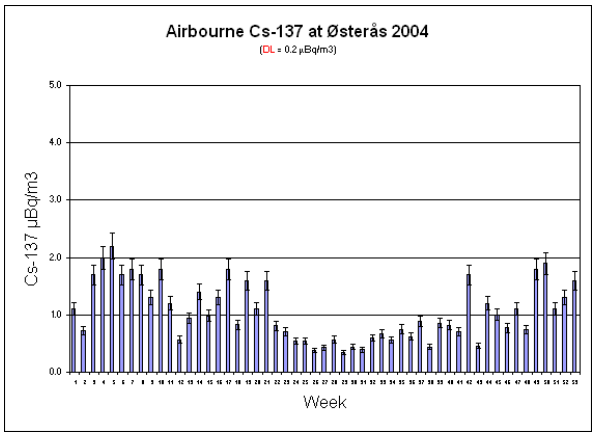

Figure 2: Airborne caesium-137 at Østerås in 2004.

In two cases we have detected radioactive nuclides from distant nuclear accidents. In 1986 there were measured large quantities of radioactive nuclides in fallout following the Chernobyl accident in the former Soviet Union. In 1993 the air monitoring station at Svanhovd measured short-lived nuclides following an accident at the reprocessing plant at Majak the same year.

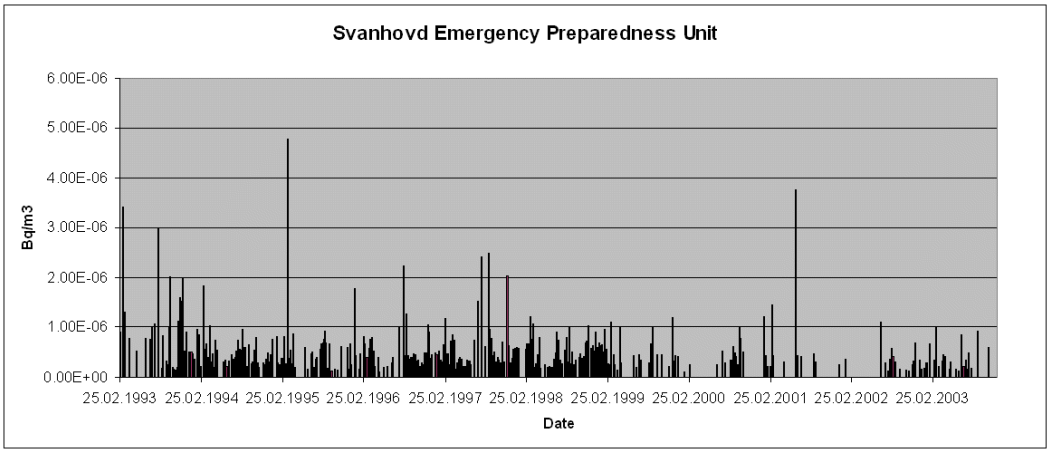

Figure 3: Airborne caesium-137 over a ten-year period from the air monitoring station at Svanhovd.

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Session V: Nuclear installations

Clearance of materials, buildings and land at the Risø site

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Abstract: During decommissioning of the nuclear facilities at the Risø site a large amount of non-radioactive materials will be cleared for use without restrictions. Furthermore, buildings and land will be cleared for non-nuclear use. This requires extensive measurements with sensitive equipment to document that clearance levels for materials, buildings and land are respected. The methods of clearing materials originating from the decommissioning of the nuclear facilities are described as well as procedures for clearing buildings and land. A low-level laboratory has been built at the Risø site for clearance measurements on materials in bulky quantities of up to 1000 kg using gamma-spectrometry. The equipment used for clearance measurements on materials will be used also for clearance measurements on buildings and land.

Introduction

The nuclear facilities to be decommissioned at the Risø site include the three research reactors DR 1, DR 2 and DR 3, the Hot Cell facility, the Fuel Fabrication facility and the Waste Management Plant with its storage facilities. All the nuclear facilities except the Waste Management Plant have been closed. The total amount of materials is expected to be cleared has been estimated to be up to about 18,000 tonnes.

Classification concerning the need for clearance measurements

For classification of potentially non-radioactive materials according to indicating their possible content of radionuclides, the history of the materials is important. From such knowledge together with preliminary measurements the materials are classified with respect to the possibility of being contaminated above the clearance levels. Similarly, areas (buildings and land) are classified according to their possibility of being contaminated above the clearance levels for such areas. Classification of areas is performed from a historical knowledge of the operation of the nuclear facilities, including routine discharges to the environment, transport of radioactive materials between the nuclear facilities on the site, results from the environmental surveillance programme and past accidents that may have caused environmental contamination.

Classification of materials: All materials are divided into to categories. The first category includes materials that could not have been in contact with radioactive materials (non-classified materials). The second category includes materials that might be contaminated with radionuclides. This second category is subdivided into two classes, class 1 and class 2 materials.

Class 1 materials are known to or assumed to be contaminated at or a little above the clearance level. These subjects have either been in direct contact with radioactive materials during operation of the facility or have been exposed to neutrons and therefore might be activated. The measurement coverage is 100% for class 1 objects, either by *in toto* measurements or measurements on samples from each subject.

Class 2 materials have a high probability of not being contaminated above the clearance level. These objects might have been in contact with radioactive materials in controlled areas within the facility. The majority of class 2 subjects will be 'clean', *i.e*. possible contamination will be far below the clearance level. The measurement coverage of class 2 subjects will be less than 100% by sampling with a strategy which will depend on the expected contamination level and the uncertainty on the measurements.

Non-classified materials have not or only to a minor extent been in contact with radioactive materials. Non-classified objects are 'clean' and control measurements are therefore not needed. Examples are here equipment and inventory in buildings at the nuclear facility used for *e.g.* offices. Anyhow, random samples will be taken from these subjects to verify that the classification is correct.

Classification of buildings and land: Areas (buildings and land) like materials are divided into two categories: areas without any reasonable potential of being contaminated are classified as non-contaminated areas. Areas that might be contaminated due to the former operation of the nuclear facilities are classified as class 1 or class 2 areas.

Class 1 areas include buildings and land with a low probability of being contaminated with 'hot spots' or being contaminated at a level above the clearance level. They include buildings at the nuclear facility in which larger quantities of radioactive materials have been handled, *e.g.* reactor buildings and land on which transport of radioactive materials between the nuclear facilities has taken place. The measurement coverage is 100% for class 1 areas by scanning and subsequent point measurements and laboratory analyses of selected samples.

Class 2 areas include buildings and land with a high probability of not being contaminated. If such areas are contaminated there is a high probability that the contamination is below the clearance level. They include auxiliary buildings at the nuclear facilities in which only minor amounts of radioactive materials have been handled and land in the immediate surroundings of these buildings. The measurement coverage of class 2 areas is 10 - 50 %, either through scanning and point measurements or laboratory analyses of selected samples.

Non-classified areas include buildings and land that have not or only to a minor extent been in contact with radioactive materials. Non-classified areas are in principle 'clean' and control measurements are therefore not needed for clearance purposes.

Clearance measurements

The nuclear authorities require documentation for the activity concentration in materials and on areas to be released. It is therefore necessary to carry out extensive control measurements for compliance with the clearance levels. The measurement strategy for materials depends on whether the materials are activated, surface contaminated and/or contaminated below the surface. Similarly, the measurement strategy for buildings and land depend on which radionuclides to be expected and their depth distribution.

Clearance laboratory and equipment for clearance measurements: A new laboratory has been built for the measurements of materials and waste being potentially non-contaminated. The laboratory is located at an adequate distance from the nuclear facilities at the Risø-site to exclude disturbing γ-radiation during decommissioning of the facilities.

Figure 1. Clearance laboratory at the Risø-site placed close to Roskilde Fjord away from the nuclear facilities. The laboratory is equipped with two germanium detectors with lead collimators.

Building materials with a low content of naturally occurring radionuclides have been used for the laboratory hall and a radon membrane has been placed below the laboratory building. The laboratory is classified as a 'clean' laboratory with changing rooms and bathrooms, to exclude activity from other areas entering the laboratory with the operators.

The laboratory hall can be equipped with up to four measuring stations as moveable concrete walls for shielding purpose can separate the hall. Presently, only two measuring stations with germanium detectors have been installed in the laboratory. The ISOCS calibration software is used to determine the content of radionuclides in the potential non-radioactive materials. The measuring stations are equipped with tables that can rotate the objects during the measurements.

Measurements on materials: Three different types of clearance measurements on materials are performed: (1) direct scanning for contamination, (2) direct measurements or laboratory analyses of selected samples and (3) *in toto* analyses. Direct measurements (*in situ* measurements) are performed either as γ-spectrometric analyses or as surface contamination measurements with contamination monitoring equipment. The number of measurements is determined from statistical considerations and evaluation of the probability of the materials being contaminated with radionuclides. *In toto* measurements also include surface-specific or volume-specific measurements.

Measurements on buildings and land: The same types of measurement as those used for materials are used for buildings and land. A combination of *in situ* γ-spectrometric measurements and laboratory analyses is always used for land allowing depth distributions of γ-emitters to be determined. Direct measurements of α-/β-contamination are often adequate for building surfaces. In laboratories where α-emitters has been used sampling from the building surfaces is necessary to determine the depth distribution of the α-activity below the surface.

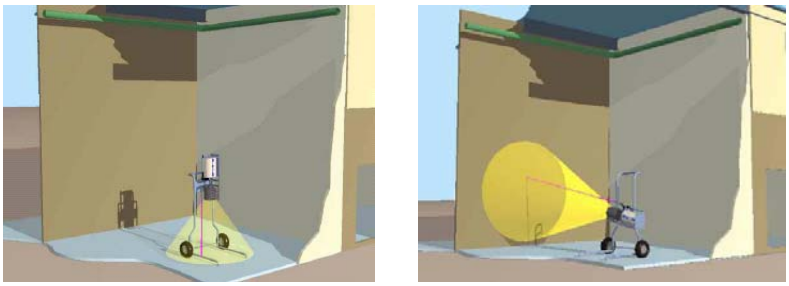

Figure 2. In situ measurements on buildings using a germanium detector with a lead collimator.

In-situ measurements include scanning (survey) of areas with mobile equipment. Several types of instruments are used for these measurements, *e.g.* hand held instruments measuring γ-dose rate or β-contamination directly on building or soil surfaces. More detailed surveys are performed by collimated germanium detectors so the nuclide-specific surface contamination is determined. The γ-spectrometric measurements are performed on building surfaces as well as on soil and asphalt surfaces as indicated in Fig. 2. The γ-spectrometric equipment used for these measurements is the same as used in the clearance laboratory.

Documentation of compliance with clearance criteria: Compliance with the clearance levels for materials and areas is documented through measurement programmes and relevant statistical tests. This assures with a high probability a correct clearance of materials and areas. The necessary number of samples and *in situ* measuring points depends on the distribution type and the selected level of significance (probability of decision error). If the contamination is not homogeneously distributed the number of samples and measuring points should be sufficient so that 'hot spots' are detected with a high probability.

Discussion and conclusions

The objective of decommissioning the nuclear facilities at the Risø site is to achieve the so-called 'Green Field' status for the buildings and land areas. This is a status where remaining buildings and areas can be used without restrictions. It implicates that all building components and equipment are either cleared or removed as radioactive waste, and that land areas cleared. Decontamination will in some cases be necessary to achieve clearance. The clearance measurements which are necessary to achieve the objective of the decommissioning of the nuclear facilities will require the use of a laboratory with a low γ-background, sensitive detectors, new types of laboratory analysis, new procedures for γ-spectra analysis and high documentation standards. Further information on measurements and sampling is written in Risø-R-1303. The principles are based on NUREG-1575.

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Radiochemical Analysis of Important Beta Radionuclides in Waste Samples for Decommissioning of Nuclear Facilities

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Decommissioning of nuclear facility

- Preparation (investigating the background radioactivity around NPP)
- Plant Cleanout
  - Removal of most radioactive component such as spend fuel elements, reactor internals, reactor vessel, etc. which is transferred for storage and disposal. (high level waste). (Evaluation of radioactivity before transferring)
- Decontamination
  - Removal of contamination from surfaces of facilities or equipment by chemical or mechanical methods, which can reduce the waste volume and active level in the waste. (Measurement of radioactivity to evaluation the decontamination, and estimation of radioactivity in the waste)
- Dismantling
  - Equipments within the facility are dismantled and classified by estimation of the radioactivity
  - Demolition and site clearance
  - Buildings demolished and radioactive wastes removed to storage or disposal facilities after estimation of the radioactivity in the waste.
- Release of the site to alternative use (measure the radioactivity level in the released area)

Waste samples and the relevant critical radionuclides for decommissioning

- Graphite (reactor)
  - ^{3}H, ^{14}C, ^{55}Fe, $^{63,59}Ni$, ^{60}Co, ^{152}Eu
- Concrete (normal or heavy)
  - ^{41}Ca, ^{60}Co, ^{55}Fe, $^{63,59}Ni$, ^{133}Ba, ^{152}Eu
- Steel/stainless steel
  - ^{55}Fe, $^{63,59}Ni$, ^{36}Cl, ^{93}Zr, ^{93}Mo, ^{94}Nb, ^{60}Co, ^{152}Eu, transuranics
- Aluminium
  - ^{60}Co, ^{63}Ni, ^{55}Fe, ^{36}Cl
- Lead
  - , ^{60}Co, ^{63}Ni, ^{55}Fe
- Water
  - ^{3}H, ^{14}C, ^{63}Ni, ^{99}Tc, ^{129}I, ^{90}Sr, ^{60}Co, ^{137}Cs, transuranics
- Ion exchange resin
  - ^{55}Fe, $^{63,59}Ni$, ^{14}C, ^{99}Tc, ^{36}Cl, ^{93}Zr, ^{93}Mo, ^{94}Nb ^{90}Sr, ^{129}I, ^{137}Cs, ^{60}Co, ^{135}Cs, transuranics

Properties of main radionuclides difficult to measure

Nuclides	Half life	Decay	Production	Measurement
^{3}H	12.33 y	β	Neutron activation of ^{2}H, Li	LSC
^{14}C	5730 y	β	Neutron activation of ^{14}N, ^{13}C, ^{17}O	LSC
^{36}Cl	3.01×10^{5} y	β	Neutron activation of ^{35}Cl	LSC
^{41}Ca	1.03×10^{5} y	EC	Neutron activation of ^{40}Ca	LSC, TIMS
^{55}Fe	2.73 y	EC	Neutron activation of ^{54}Fe	LSC
^{63}Ni	100 y	β	Neutron activation of ^{62}Ni	LSC, AMS
^{59}Ni	7.6×10^{4} y	EC	Neutron activation of ^{58}Ni	LSC. AMS
^{90}Sr	28.8 y	β	Fission product	LSC, beta counting
^{94}Nb	2.03×10^{4} y	β-, γ	Fission product	LSC, γ-spectrometry
^{99}Tc	2.11×10^{5} y	β-	Fission product	LSC, ICP-MS
^{129}I	1.57×10^{6} y	β-	Fission product	LSC, beta counting NAA
^{93}Zr	1.53×10^{6} y	β-	Neutron activation of ^{92}Zr	LSC
^{93}Mo	4.0×10^{3} y	EC	Neutron activation of ^{92}No	LSC, AMS
^{239}Pu	2.4×10^{4} y	α	Neutron capture of ^{238}U	α spectrumetry, ICPMS
^{240}Pu	6.56×10^{3} y	α	Neutron capture of ^{239}Pu	α spectrumetry, ICPMS
^{241}Pu	14.35	β-	Neutron capture of ^{240}Pu	LSC
^{241}Am	432.2 y	α	Decay from ^{241}Pu	α, γ-spectrometry
^{237}Np	2.14×10^{6} y	α	Neutron capture of ^{236}U	α spectrumetry, ICPMS
^{243}Cm	29.1 y	α	Neutron capture and decay	α-Spectrumetry

Rapid analysis of ^{3}H and ^{14}C in graphite, concrete

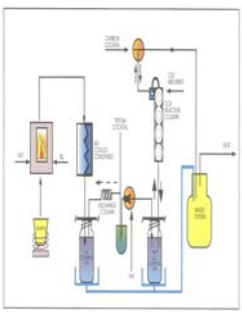
- *Production of ^{3}H in reactor*
  - $^{2}H(n, \gamma)^{3}H$
  - $^{6}Li(n, \alpha)^{3}H$
  - $^{3}He(n, p)^{3}H$
- *Production of ^{14}C in reactor*
  - $^{13}C(n, \gamma)^{14}C$
  - $^{14}N(n, p)^{14}C$
  - $^{17}O(n, \alpha)^{14}C$

Nuclides	Half life	Decay	Energy, keV	Measurement
^{3}H	12.35 y	β	18.6	LSC
^{14}C	5736 y	β	156	LSC

Measurement of C-14 and H-3

- Low energy
  Liquid Scintilation Counter
- Continueous epectrum
  Separation before measurement

Rapid separation of ^{3}H and ^{14}C waste samples by combustion using Packard Oxidizer

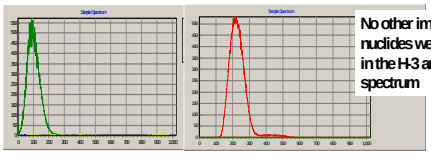


Decomposition of graphite and concrete for release C-14 and H-3 using Oxidizer

sample	weight	^{3}H added	^{14}C added	^{3}H counts	^{14}C counts	Recovery or memory of ^{3}H	Recovery or memory of ^{14}C
	mg	Bq	Bq	CPM	CPM	%	%
graphite	1060.3	0	0	10	12		
graphite	55.95	0	0	16	13		
graphite	41.2	0	0.47	11	13925	0.01	94.09
graphite	30.03	0	1.46	25	60360	0.02	105.50
graphite	34.76	0.4	0	12083	20	96.21	0.16
graphite	39.27	4.87	0	115088	24	93.07	0.02
concrete	35.23	0	0.47	16	12292	0.10	97.87
concrete	41.57	0	2.92	27	119809	0.02	96.93
concrete	38.35	0.4	0	12338	26	98.24	0.2
concrete	37.63	4.87	0	124118	28	100.42	0.02

Analysis using different method

	C-14 activity (Bq/g)					Tritium activity (Bq/g)				
DR2 graphite	Oxidizer		Acid digestion		Differenc	Oxidizer		Acid digestion		Differenc
	Average	SD%	Average	SD%	%	Average	SD%	Average	SD%	%
TK55 Yi	212	45	241	89	136	4697	173	5223	89	112
TK55 Yy	7551	29	7256	68	39	197488	115	178081	123	99
TK75 li	14597	28	15236	59	44	981184	55	996521	98	71
TK75 ly	6769	16	6257	81	76	343722	97	353864	88	29



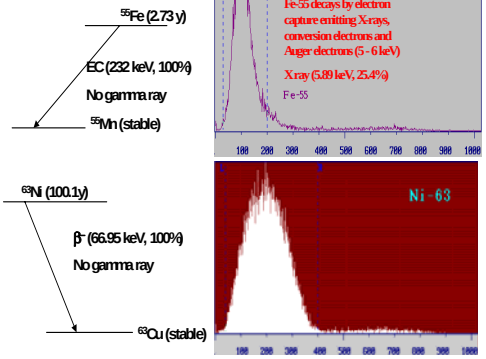
No other impurity nuclides were seen in the H-3 and C-14 spectrum

Features of new method for ^{3}H and ^{14}C in soild samples (concrete, graphite, resin, wood, paint, etc.)

- Short analytical time (2-3 min. preparation time per sample), while the traditional method need 3-4 hours.
- Good accuracy, (Confirmed by standard addition method and comparing with acid digestion method).
- Satisfactory sensitivity (the detection limits of ^{14}C and H-3 are 0.7 and 0.5 Bq/g, respectively).

Hou X.L., Applied Radiation and Isotopes, 62:871-882, 2005,

Decay of ^{63}Ni and ^{55}Fe



Analytical method for ^{63}Ni and ^{55}Fe

- Due to their low energy of beta particle and measurable electrons, LSC is the most suitable method for their measurement.
- Due to their pure beta and EC decay, they have to be separated from matrix elements and all other radionuclides.

Mearement methods

^{55}Fe: X-ray spectrometry (<1%);
LSC (30-45%)

^{63}Ni: gass flow counting(anti-coincidence, <10-50%)
Ion implanted silicon detector (1-6%)
LSC (60-80%)

- Analytical procedure:
  - Decomposition of sample
  - Separation of Ni or Fe from matrix elements and all other radionuclides
  - Preparation of a suitable solution for LSC measurement.

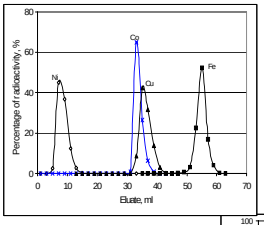
Interfering Radionuclides

Nuclide	Half-life	Decay	Nuclide	Half-life	Decay
^{60}Co	5.27 y	β^-,γ	^{3}H	12.33 y	β^-
^{58}Co	70.86 d	β^+,γ	^{14}C	5730 y	β^-
^{152}Eu	13.54 y	$\varepsilon,\beta^-,\gamma$	^{133}Ba	10.51 y	ε
^{154}Eu	8.59 y	β^-,γ	^{41}Ca	1.03E5 y	ε
^{51}Cr	27.7 d	ε,γ	^{36}Cl	3.01E5 y	β^-,ε
^{65}Zn	244.3 d	$\varepsilon,\beta^+,\gamma$	^{137}Cs	30.7 y	β^-
^{54}Mn	312.3 d	$\varepsilon,\beta^+,\gamma$	^{134}Cs	2.06 y	β^-,ε
^{151}Sm	90 y	β^-	^{90}Y	64 h	β^-
^{90}Sr	28.79 y	β^-			

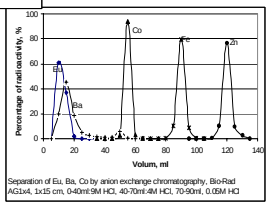
Methods for separation of Ni and Fe

- Precipitation as $Ni(OH)_2$ and $Fe(OH)_3$ to separate from Sr, Cs, ^{3}H, ^{14}C, Ba, Ca, Cl.
- Ion exchange to separate Ni, Fe from Co, Cu, Zn, transuranics, and each other
- Extraction chromatography using Ni-column with dimethylglyoxime (DMG) to separate and purify Ni from Cr, Co, Cu, Fe, Eu, etc.

Separation of Ni, Co, Eu, Ba by anion exchange chromatography

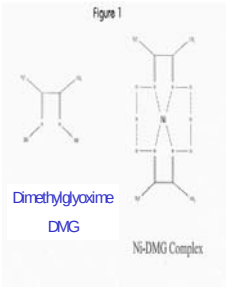


- loading at 9 mol/l HCl, Ni is not absorbed on column, while others are absorbed.
- Removing Co, Cu by washing with 4 mol/l HCl
- Eluting Fe with 0.5 mol/l HCl



Separation of Eu, Ba, Co by anion exchange chromatography, Bio-Rad AG1x4, 1x15 cm; 0-40ml:9M HCl, 40-70ml:4M HCl, 70-90ml, 0.05M HCl

Separation of Ni using Ni-DMG complex

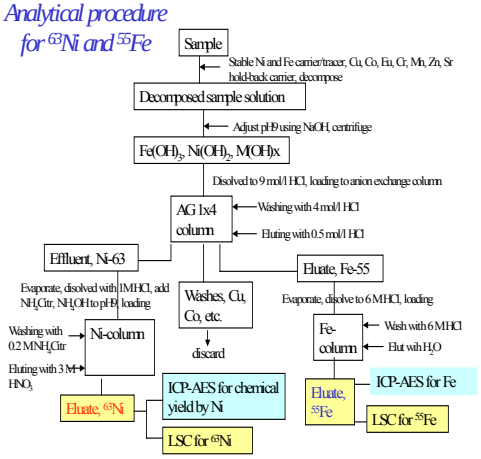


Purification of Ni by specific Ni-extraction chromatography

Element	Recovery or decontamination factor
Ni^{2+}	>98.5%
Fe^{3+}	10^4
Co^{2+}	10^3
Ba^{2+}	10^4
Eu^{3+}	10^4
Cs^{+}	10^4
Sr^{2+}	10^4

- Ni specific extraction chromatography has a higher decontamination to most of elements, such as Fe, Co, Cu, Cr. Mn, Ba, Eu, transuranics, etc.
- A higher recovery of Ni can be obtained in the procedure.

Analytical procedure for ^{63}Ni and ^{55}Fe

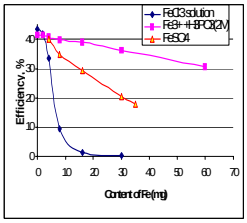


The recovery of Fe and Ni and decontamination factors for main interfering radionuclides

Interference	Recovery/decontamination factor		Interference	Recovery/decontamination factor	
	Fe fraction	Ni fraction		Fe fraction	Ni fraction
^{55}Fe	85-95%	$>10^5$	^{133}Ba	$>10^6$	$>10^6$
^{63}Ni	$>10^5$	80-95%	^{134,137}Cs	$>10^6$	$>10^6$
^{58,60}Co	$>10^5$	$>10^5$	^{89,90}Sr	$>10^6$	$>10^6$
^{152,154}Eu	$>10^6$	$>10^5$	^{41,45}Ca	$>10^6$	$>10^6$
^{151}Sm	$>10^6$	$>10^5$	^{36}Cl	$>10^6$	$>10^6$
^{54}Mn	$>10^5$	$>10^6$	^{3}H	$>10^6$	$>10^6$
^{51}Cr	$>10^6$	$>10^5$	^{14}C	$>10^6$	$>10^6$

For all interferring radionuclides, the decontamination factors higher than 10^5.

Preparation of separated ^{55}Fe for LSC



- Yellow colour Fe^{3+} is a very effective quenching agent
- Reduction of Fe^{3+} to Fe^{2+} can reduce the quench, but Fe^{2+} is not stable and Fe^{2+} also has some colour quench.
- In H_3PO_4 solution, a stable and colourless Fe- H_3PO_4 complex can be formed, therefore can significantly reduce the Fe^{3+} colour quench.
- As high as 40% counting efficiency of ^{55}Fe was obtained.

Features of the method for ^{55}Fe and ^{63}Ni

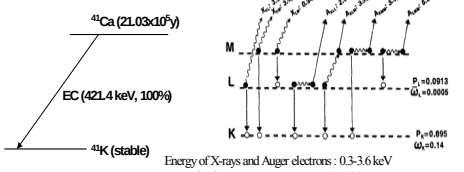
- The recoveries of Fe and Ni are 80-95%
- The decontamination factors for most of interferring nuclides are higher than 10^5 .
- The detection limit for ^{63}Ni and ^{55}Fe are 0.015 Bq and 0.035 Bq respectively.
- For the determination of ^{59}Ni, the separated and purified Ni can be electroplated on stainless steel disk and measuring its x-rays of 6.9 keV using low-energy gamma detector.

Hou, Østergaard, Nielsen, Anal. Chim. Acta, 535:297-307, 2005,

^{41}Ca in the concrete

Activation products of calcium isotopes

Nuclide	Target isotope Abundance %	Reaction	Cross section, bar	Half life	Decay
^{41}Ca	96.94	^{40}Ca(n, γ)^{41}Ca	0.41	1.03×10^5 y	EC
^{45}Ca	2.086	^{44}Ca(n, γ)^{45}Ca	0.84	162.7 d	β-
^{47}Ca	0.004	^{46}Ca(n, γ)^{47}Ca	0.7	4.54 d	β, γ
^{49}Ca	0.187	^{48}Ca(n, γ)^{49}Ca	1.0	8.72 min.	β, γ



Energy of X-rays and Auger electrons : 0.3-3.6 keV
Determination: X-ray spectrometry (<0.08%)
LSC (10-20%)

Determination ^{41}Ca in heavy concrete

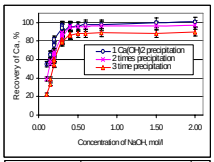
- Separation fro matrix
  - Decomposition of heavy concrete by alkali fussion
  - Leaching Ca by acids
- Separation from active metals such as ^{60}Co, ^{152}Eu, ^{55}Fe, ^{63}Ni, ^{65}Zn, ^{54}Mn, ^{51}Cr,etc.
  - Precipitation with $Fe(OH)_3$ by hydroxides at pH9
- Separation from other alkaline metals, such as ^{133}Ba, ^{226}Ra and ^{90}Sr.
  - $BaCrO_4$ and $SrCrO_4$ precipitation
  - $BaCl_2$ and $SrCl_2$ precipitation in HCl solution
  - $Ca(OH)_2$ precipitation in NaOH solution

Concrete in the reactor

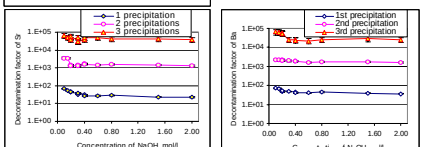
- **Ordinary concrete**
- **Heavy concrete (50-70% $BaSO_4$ was added)**

The new method for the separation of Ca from Sr and Ba

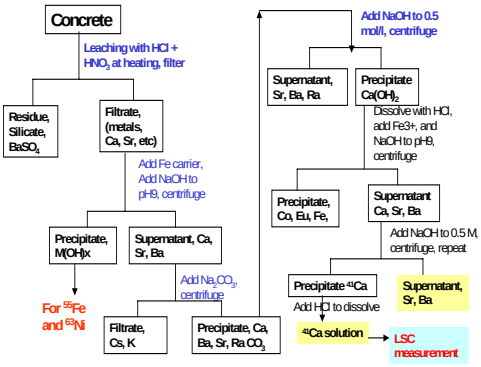
- **Separation of Sr from Ca by $Ca(OH)_2$ precipitation**
  - **$Ca(OH)_2$: insoluble, Ksp = 5.2×10^{-6}**
  - **$Sr(OH)_2$ and $Ba(OH)_2$: Soluble in alkine solution**



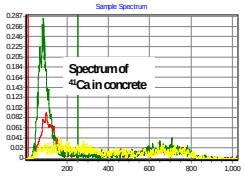
Precipitate Ca as $Ca(OH)_2$ at 0.5 – 0.8 mol/l NaOH, repeat 3 times, 85% Ca can be recovered, and the decontamination factor for Sr and Ba are higer than $5x10^4$

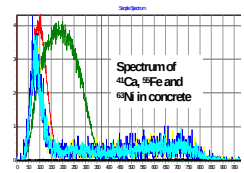


Procedure for simultaneous determination of ^{41}Ca, ^{55}Fe and ^{63}Ni



Spectra of ^{41}Ca in heavy concrete from DR-2

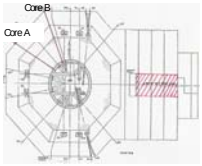




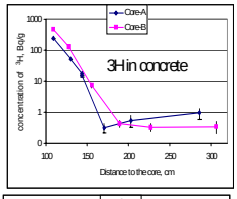
- Features of Method for ^{41}Ca
  - A separation of ^{41}Ca from concrete is easy to operate
  - Good decontamination from interferring radionuclides ($>10^4$)
  - The chemical yields of the separation procedures for ^{41}Ca is 80-90%
  - The detection limits for ^{41}Ca is 0.020 Bq.
  - Simultaneous determination of ^{41}Ca, ^{90}Sr, ^{55}Fe and ^{63}Ni.

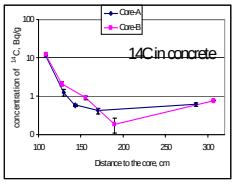
Hou, Radiochim. Acta. in press, 2005

Results of ^{3}H, ^{14}C, in graphite and concrete from Danish reactor, DR-2



Sampling of concrete and graphite from danish reactor, DR-2

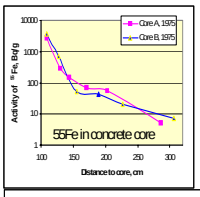


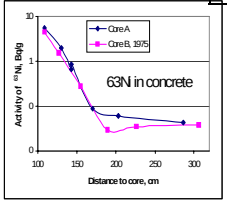


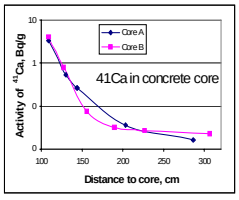
Graphite from DR-2

	^{3}H, Bq/g		^{14}C, Bq/g	
Code	Average	SD %	Average	SD %
TK5.5 Yi	2.12E+02	4.50	4.70E+03	17.33
TK5.5 Yy	7.55E+03	2.85	1.97E+05	11.48
TK5.5 Ii	1.96E+04	2.43	1.04E+06	10.36
TK5.5 Iy	9.04E+03	3.01	4.31E+05	0.93
TK7.5 Yi	2.67E+04	2.19	1.11E+06	3.39
TK7.5 Yy	1.08E+04	1.74	3.67E+05	7.88
TK7.5 Ii	1.46E+04	2.76	9.31E+05	5.53
TK7.5 Iy	6.77E+03	1.61	3.44E+05	9.66
Sample-G	3.21E+03	1.25	7.16E+04	11.73

Results of ^{55}Fe, ^{63}Ni and ^{41}Ca in concrete from danish reactor, DR-2







^{55}Fe and ^{63}Ni in graphite of DR-2

Sample No	^{55}Fe		^{63}Ni	
	Recovery, %	Bq/g	Recovery, %	Bq/g
DR-3-T	92.2	545000	94.63	5552
Iy7.5	90.4	0.53	93.89	92.5
Iy5.5	90.6	1.05	93.74	22.3
Yi7.5	92.5	1.92	93.35	7.71
Yi5.5	91.3	9.21	91.56	43.1

Regulatory requirements for radiation safety in the design of a new Finnish NPP

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Abstract: STUK reviewed the electricity generating company TVO's application for the Construction License of the fifth Finnish nuclear power reactor in 2004-2005. Based on this review STUK prepared its statement on safety together with a safety assessment report of the new plant to the Government. By virtue of the Nuclear Energy Act (990/87) and the Council of State Decision (395/91) on General Regulations for the Safety of Nuclear Power Plants, Radiation and Nuclear Safety Authority (STUK) issues detailed regulations, YVL-guides, concerning the safety of nuclear power plants. Several YVL Guides deal with radiation safety (site, abatement of releases, worker radiation protection, emergency arrangements etc). This paper discusses some radiation safety related requirements in the design of a new Finnish NPP and their implementation in the construction license documents and procedures.

Introduction

Two operating nuclear power plants in Finland with two BWR units at Olkiluoto site and two PWR units at Loviisa site were commissioned between 1977 and 1981. The licensing process of a new nuclear power plant in Finland is shown in Figure 1. The project of the fifth Finnish nuclear power reactor was formally started in May 1998 with Environmental Impact Assessment (EIA) process. Then the electricity generating company TVO submitted the application for a Decision in Principle in November 2000. The Finnish Government made the decision in January 2002, which Parliament ratified in May 2002.

In 2003, TVO proposed the plant site to be Olkiluoto and made a contract with a consortium of Framatome ANP and Siemens AG to build an EPR (European Pressurised Water Reactor). TVO submitted the application for Construction License in the beginning of 2004 and the license was granted by the Government in February 2005. Based on TVO's schedule, estimated construction time is about four years. The Operating License evaluation process takes approximately one year. Thus, the new unit could be in full power operation in 2009 if no major delays occur.

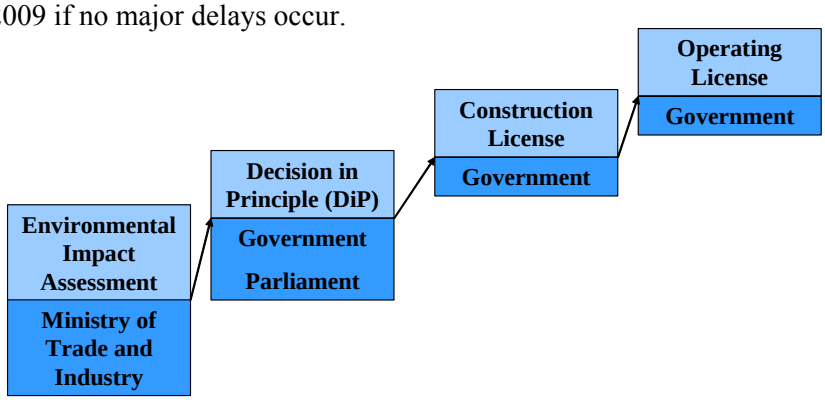

Figure 1. The licensing process of a new nuclear power plant in Finland.

Construction License Application Review at STUK

At the same time as the application for Construction License was sent to the Ministry of Trade and Industry, TVO submitted the required licensing documentation to the Radiation and Nuclear Safety Authority (STUK). According to the Finnish Nuclear Energy Decree Section 35, these documents include:

- Preliminary Safety Analysis Report (PSAR)
- Proposal for a Classification Document
- Description of Quality Assurance during Construction
- Preliminary plans for Physical Protection, Emergency Preparedness, and Safeguards
- Applicant's Arrangements for the Regulatory Review by STUK
- Other reports that STUK considers necessary (design phase PSA).

Based on the review of these documents, STUK prepared its statement on safety and a safety assessment report, which were submitted to the Ministry of Trade and Industry in January 2005. STUK's positive statement on safety was a prerequisite for the Government to grant the Construction License. In the statement, STUK indicated specific observations on and some further demands for the plant safety and protection.

Radiation Safety in YVL Guides

The guide YVL 7.18, Radiation safety aspects in the design of NPPs, was updated in 2003. In this guide, e.g. accident situations including severe accidents and aspects of decommissioning of the plant were taken into account more in detail. Other relevant radiation guides during the construction license review were:

- YVL 1.10 Safety criteria for siting a NPP
- YVL 7.1 Limitation of public exposure in the environment of and limitation of radioactive releases from NPPs
- YVL 7.2 Assessment of radiation doses to the population in the environment of a NPP
- YVL 7.3 Calculation of the dispersion of radioactive releases from a NPP
- YVL 7.5 Meteorological measurements at NPPs
- YVL 7.6 Monitoring of discharges of radioactive substances from NPPs
- YVL 7.11 Radiation monitoring systems and equipment in NPPs.

Further relevant guides for the operating license review are:

- YVL 7.4 NPP emergency preparedness
- YVL 7.7 Radiation monitoring in the environment of NPPs
- YVL 7.8 Environmental radiation safety reporting of NPPs
- YVL 7.9 Radiation protection of NPP workers
- YVL 7.10 Monitoring of occupational exposure at NPPs. [1]

Collective Dose of NPP Employees (ALARA)

In the updated regulatory guide YVL 7.18, a design upper limit for an annual personnel collective dose of 0.5 manSv per 1 GW of net electric power averaged over the plant life is set forth. In the European Utility Requirements (EUR) document, the target for annual collective effective dose averaged over the plant life is set as 0.5 manSv per reactor unit.

Average personnel collective radiation doses per reactor for operating OECD country NPPs [2] and for existing Finnish NPPs for the years 1991-2001 are shown in Figure 4.

The collective dose at the Olkiluoto NPP has been clearly under the international reference values of the BWR reactors. On the other hand, the comparison of the collective dose at the Loviisa NPP to the PWR reactors does not more give such an excellent result. Average collective doses per reactor of the German Konvoi generation NPPs (Emsland 1, Isar 2 and Neckarwestheim 2) and French N4 generation NPPs (Chooz B1 and B2, statistics only from the year 2001) [2] are also shown in Figure 2. The statistics of the Konvoi NPPs would indicate that the collective dose in the EPR could be low.

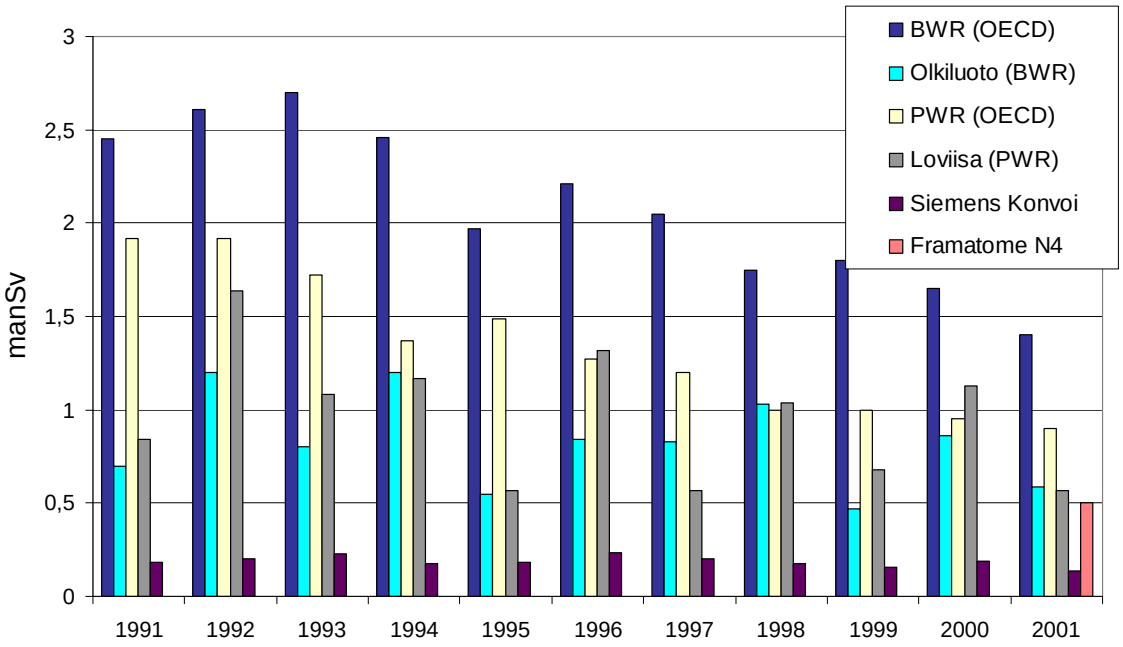

Figure 2. Average personnel collective radiation doses per reactor

The PSAR review of STUK experts on plant radiation sources, shielding, lay-out and radiation protection arrangements, as well a topical inspection made to the FANP by STUK pointed out some aspects where the applicant and the wendor (designer) shall enhance efficient communication between different expert and designer groups.

Accident Situation and On-Site Radiation Safety

In a nuclear power plant, on-site habitability during accident situations has to be taken into account. “Habitability” determines conditions whether or not the occupancy of a certain area inside or outside the site buildings is possible on a continuous or transient basis. The regulatory guide YVL 7.18 requires analyses of the magnitude and location of the possible radiation sources and evaluation of doses in different accident management and emergency preparedness measures. In the design process, these doses shall not exceed the normal dose limits of a radiation worker.

Assessment of the on-site habitability during severe accidents at the existing Finnish nuclear power plants has been primarily done during 1980’s and 1990’s. A reassessment was done in 2002-2004 [3]. Radiation hazard was classified into three parts, i.e., possible direct radiation from the containment, air contamination and systems carrying radioactive air or water. The results showed that direct radiation from the containment is generally adequately shielded but penetrations and hatches have to be separately analysed and the radiation dose levels near them might become rather high. An interesting result was that air contamination also in the building next to the containment might be a hazard even if

the containment is intact and leaks only at the nominal rate. Systems outside the containment can also create higher local radiation levels, e.g. near the emergency core cooling systems, containment spray system, sampling systems and containment filtered venting system.

Assessment made by STUK based on PSAR of Olkiluoto 3 PSAR revealed some spots where adequate shielding and lay-out arrangement in the design were needed.

Minimising Discharges of Radioactive Materials

The reactor, systems and components containing radioactive substances shall be designed in such a way that releases of radioactive substances and the radiation exposure of the population living in the vicinity of the plant can be kept low. Systems which are capable of cleaning fluids and gases containing radioactive substances shall be effectively limit radioactive releases. Radioactive releases from a nuclear power plant during normal operation are, to a great extent, determined by leakage from the nuclear reactor fuel rods, reactor coolant and its fission and corrosion products, maintenance operations and waste management (including purification and retention of exhaust gases and liquids).

Based on PSAR review made by STUK, due consideration has been given to the minimization of discharges of Olkiluoto 3 reactor plant. Over the past years fuel leaks at the German and French reference plants have been minor. The same is expected to apply to Olkiluoto 3 reactor fuel. The materials for the reactor cooling circuit have been selected and the water chemistry designed with a view to minimizing the creation of radioactive corrosion products. The reactor coolant purification system, the processing system for gaseous wastes, the storage and processing system for liquid wastes and the processing system for radioactive concentrates will be based on technology used at the German reference plants, with improvements based on operating experience. The purpose of these systems is to limit the releases of radioactive materials into the environment. These systems are designed with due regard to the Best Available Techniques (BAT). Adequate exhaust air filters have been designed for installation in the ventilation systems.

Severe Accident Mitigation, Other Issues

At Olkiluoto 3 reactor plant severe accident management against major core damage will be based on proper containment functions with a unique core catcher. These measures will prevent any major release of radioactive materials (Cs - 137 release in excess of 100 TBq shall have a probability less than $5\ 10^{-7}$ 1/a). In addition, the Olkiluoto 3 plant will be constructed e.g. according to the design features which provide safety and prevent significant releases even in the case of a crash of a big passenger aircraft.

The plant radiation monitoring systems description was a proper part of the PSAR description. STUK review indicated that the system design covered well the main requirements of YVL 7.11.

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[1] http://www.stuk.fi/english/publications/yvl-guides.html

[2] MADRAS database, version 3.8 (Rev. 9), European Technical Centre (CEPN), 2001/2002.

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Discharges of radioactivity from Institute for Atomic Energy in the 1950s and 1960s

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Abstract: In August 2004 the local population living near Institute for Energy Technology (former Institute for Atomic Energy) in Norway expressed the concern that cancer incidences in their family and among their neighbours were caused by discharges of radioactivity from Institute for Atomic Energy (IFA) in the 1950s and 1960s. Based on this article and the following series of newspaper articles, Institute for Energy Technology (IFE) was requested by Norwegian Radiation Protection Authority (NRPA) to give detailed information of discharges to the atmosphere and to the nearby lagoon (Sogna) specifically in the period from 1948 to 1964, but also for the period 1964 to 1978. This presentation gives an overview of the results from our archive research of the discharges in the period 1948 to 1967 from Institute for Atomic Energy.

Introduction

In August 2004 the local newspaper ”Romerikes Blad” presented an article where some people living near the lagoon Sogna in the 1959s and 1960s expressed their concern that cancer incidences in their family and among their neighbourhood could be caused by discharges of low level radioactive waste from Institute for Atomic Energy (IFA, now called Institute for Energy Technology, IFE) at Kjeller, Norway, to this lagoon. In the following months the local population, health authorities, politicians and ”instant experts” speculated and commented on the cause of the cancer incidences almost every day. Individuals in the local population claimed that they frequently had bathed and played in the lagoon Sogna and in the sewage outlet despite explicit advices against this from the local health authorities. A need for a health examination of the population was advocated.

In a letter to IFE, dated 10 September 2004, the Norwegian Radiation Protection Authority (NRPA) asked IFE to specify nuclides and radioactivity levels discharged to water and to the atmosphere in more detail in the period 1948-1963 than previously reported [1] and that more detailed information for the period 1964-1978 also was needed. A deadline of 1 November 2004 that later was extended to 15. November 2004 was given for submitting the information.

Based on the request from NRPA IFE carried out an extensive investigation of old archive materials in order to try to dig out more detailed information of discharges than had previously been available. The report IFE/KR/F-2004/194 [2] containing available information was submitted to NRPA on 15 November last year.

Bases on this report the NRPA presented their conclusion regarding health consequences for people living near IFA to the Ministry of Health and Social Affairs and gave their advice on the expediency of a health examination of the population and how this could be carried out [3].

A short history activities and discharges of from IFA up to 1967

Discharges of low level liquid radioactive waste

The Institute for Atomic Energy was established in 1948. Operation of the first research reactor, the 0,5 MW Jeep I, was started in 1951 and was in operation up to 1966. The production of radiopharmaceuticals started in 1951/1952. In the 1950s only radioactive medical products based on

^{131}I and ^{32}P are reported. The Chemical Department was organised in 1952 and worked in the same building that housed the Isotope Laboratory for radiopharmaceutical production.

In the investigation of old archive materials no documents or reports have been found that describe any discharge in the period 1948 to mid 1954. In a letter to "Statens Radiologisk-Fysiske Laboratorium" (SRFL, now Norwegian Radiation Protection Authority, NRPA) IFA asked if they had any objection to discharges of liquid radioactive waste containing specified weekly maximum activities of short lived nuclides from the Isotope Laboratory to the nearby river Nitelva though the ordinary sewage drain from IFA. In the their answer SRFL said that they had no objections and this was considered by IFA as a permission for discharges of low level liquid radioactive waste containing short lives radioactivity to the lagoon Sogna which was connected to Nitelva. Based on this it can be assumed that the discharges of liquid radioactive waste started in the mid 1954, but no documents have been found that give detailed information of discharge frequency or activity levels. In letters from IFA in the period 1954 to 1957 the expected maximum levels of discharges per week of radioactivity are described. These are given in Table 1.

Unfortunately the SRFL was not the right institution to apply for discharges to the Nitelva, and according to the Norwegian legislation applications for discharges to river systems should be directed to "Norges Vassdrags og Elekrisitetsvesen" (NVE). In correspondence with NVE and SRFL in 1956 and 1957 IFA repeated it's application for discharges to Nitelva River, but permission was not given, and some institutions advised against discharges of radioactivity. IFA therefore stopped discharges of liquid radioactive waste to Sogna in May 1957. The liquid waste containing ^{131}I, ^{127}Te, ^{129}Te and ^{32}P was collected in a 10 m^3 tank where also liquid waste containing fission products was collected from operation of the Chemical Department.

The application for discharges and consequences was thoroughly evaluated by SRFL in 1957 and all relevant information was submitted from NVE to the Ministry for Industry in 1958. In 1958 the Ministry refused the application but would reassess it in connection with operation of the planned Radioactive Waste Treatment Plant. In the period from May 1957 to September 1961 all liquid radioactive waste was collected in the 10 m^3 tank and no releases to the Sogna lagoon occurred.

In 1961 the Isotope Laboratory moved their production to a new building equipped with tanks for collecting liquid radioactive waste and increased their product spectre and volumes in the following years. A new Uranium Reprocessing Pilot Plant started test-operation in 1959 and ordinary operation in 1961 in a separate building. In 1961/1962 the Chemical Department moved their operation to this building. The new Radioactive Waste Treatment Plant started test-operations in 1961/1962 in a building linked to the Uranium Reprocessing Pilot Plant.

It soon became clear that the tanks containing radioactive waste from the Isotope Laboratory and from the Uranium Reprocessing Pilot Plant (URPP) and Chemical Department were going to be filled up. Based on close communication between IFA and SRFL in this matter, oral permission was granted in 1961 to empty these tanks though the ordinary sewage drain ending in the Sogna lagoon, provided that measurements of the content were made before discharges. Measurements were performed and reported as total activity and in grams for uranium. Measurements of the activity of separate nuclides were not performed. The discharges in from mid 1961 to 1962 are given in Table 2.

In June 1962 IFA again applied for permission to discharge low level radioactive wastewater to the Nitelva. In the application the maximum discharge was specified as 2 curie liquid radioactivity in periods of 30 days. The application was sent to "Det Kgl. Departement for Industri og Håndverk". The 3 December 1963 IFA received permission for discharges in accordance with the application, but restricted to 0,5 curie in periods of 30 days as long the old sewage drain was used (see below) and with restrictions when the flow of water in the river fell below 1 m^3/sec. To follow up the discharge limit given in the permission a concept called curie-equivalent (Ci-e) was used based on recommendation from the International Commission on Radiological Protection. The activity levels in the discharges were measured as gross β- and gross α-activity and are given in Table 3 for the period 1963 to May 1967.

Pipelines used for discharges of liquid radioactive waste

In the periods 1954 to 1957 and 1961 to 1967 liquid radioactive waste were discharged though the ordinary sewage drain from the Kjeller area to the Sogna lagoon. The Sogna lagoon was a branch of the Nitelva and had a rather low water flow in some periods of the year. The permission for discharges though the sewage drain therefore contained regulations linked to the water flow.

Planning of a new pipeline for discharges of low radioactive wastewater from IFA started in 1961. This pipeline from IFA to the Sogna lagoon was finished in 1963 but was not taken into operation. In order to ensure a high enough flow of water at the outlet, the pipeline was extended 900 meter from the Sogna lagoon into Nitelva, following the river downstream to where it meets another river. Discharges of low level radioactive waste water though this pipeline was started 15 May 1967 and has been going on since except for a major maintenance and change of outlet location quite recently.

Discharges to the atmosphere up to 1967

From the beginning of operations of the nuclear facilities at IFA health physics and radiation protection work were focussed on personal dosimetry and work conditions inside the buildings. It was well known that irradiation of air by neutrons produces ^{41}Ar and that this noble gas was released to the atmosphere during operation of the Jeep I reactor. Calculation performed in 1958 estimated a discharge of 1 Ci (37 GBq) of ^{41}Ar per hour of operation when the reactor was operated at 450 kW. In the period June 1956 to 1966 the average discharge of ^{41}Ar per year was approximately 200 TBq, varying with the total hours of operation per year.

The first measurements of discharges of iodine from the Isotope Laboratory to the atmosphere are given in a separate report from the Health Physics Department in 1961. The measurements were given in units of $\mu\mu Ci/m^3$ in the off gas from the laboratory. Measurements of concentrations of iodine in off gas from the Isotope Laboratory and of gross β and ^{89}Sr + ^{90}Sr in the off gas from the Uranium Reprocessing Pilot Plant were reported in the health physics reports from 1962 to 1965. No reported measurements can be found for 1966 and 1967, as but from 1968 total discharges to the atmosphere are reported in annual reports. The results of the measurements in 1961-1965 are given in Table 4 as the average radioactivity per m^3 in off gas from these laboratories per year. Knowledge of the total capacity of the ventilation systems is lacking.

Evaluations and conclusions given by the Norwegian Radiation Protection Authority

The 15 November 2004 IFE submitted its report IFE/KR/F-2004/194 [2] to NRPA. This report contained all available information regarding discharges and environmental monitoring in the period 1948 to 1963 and 1964 to 1978. The report also included references to annual reports previously submitted to NRPA, describing discharges from 1961 up to 1978.

Based on this information the NRPA calculated doses to the population living in the proximity of IFA. The results are reported in their report “Strålevern Rapport 2005:3” [3]. In order to perform the dose calculation NRPA had to make a series of assumptions regarding discharges of nuclides and activity levels not reported in the available documents and distributions of nuclides in the gross β- and gross α measurements. Assumptions of the behaviour of people living near the Sogna lagoon regarding bathing, playing in the outlet of the sewage drain and consumption of fish caught in the lagoon or river were also made. Many of there assumptions are very conservative, and the real doses can be assumed to be considerably lower than the calculated ones.

NRPA reported that the dose calculations showed doses up to 1,5 mSv to individuals in the population in single years, and that the discharges in 1963 could have resulted in doses in excess of 2 mSv. From this results, and compared to accumulated doses from the natural background radiation, the NRPA

draw the conclusion that the probability of cancer induction from discharges of radioactivity from IFA/IFE and from other enterprises in the Kjeller area has been and is very small.

Table 1. Estimated maximum limit of weekly discharges of ^{131}I, $^{127,129}Te$ and ^{32}P in wastewater in 1954-1957

	Estimated maximum limit for weekly discharges of				
Nuclide Reference	^{131}I (GBq)	^{127}Te (GBq)	^{129}Te (GBq)	^{32}P (GBq)	Volume (l)
Estimate April 1954	7,4	3,7	3,7	-	
Estimate May 1956	3,7	3,7	3,7	-	
Estimate October 1956	3,7	3,7	3,7	0,037	30
Estimate April 1957	11	3,7	3,7	0,037	30

Table 2. Total activity discharged in low radioactive wastewater in 1961 and 1962

	Total activity in discharged wastewater from the		
Year	Isotope Laboratory (MBq)	Radioactive Waste Treatment Plant (MBq)	Uranium (g)
1961	5 500	0	11
1962	6 616	167	0

Table 3. Gross β- and gross α-radioactivity discharged in wastewater from 1963 to 15 May 1967

	Radioactivity in discharged wastewater from the				
Year	Isotope Laboratory	Radioactive Waste Treatment Plant			
	Gross β (MBq)	Gross β (MBq)	Gross α (MBq)	^{90}Sr(measured) (MBq)	Uranium (g)
1963	2 283	1 345	9,8		861
1964	17 267	5 225	13,6	12,4	
1965	3 913	3 302	73,3	15,8	
1966	13 940	1 614	54,5		
1967 (to 15 May)	808	224	5,5		

Table 4. Average radioactivity concentration in off gas in 1961-1965

	In off gas Isotope Laboratory	In off gas from Uranium Reprocessing Pilot Plant	
Year	^{131}I (Bq/m^3)	Gross β (Bq/m^3)	$^{89}Sr + ^{90}Sr$ (Bq/m^3)
1961	142		
1962	9,3	$1{,}1 \cdot 10^{-2}$	$1{,}6 \cdot 10^{-3}$
1963	3 583[1)]	$1{,}2 \cdot 10^{-2}$	$1{,}0 \cdot 10^{-3}$
1964	522	$3{,}8 \cdot 10^{-3}$	$2{,}0 \cdot 10^{-4}$
1965	457	$3{,}1 \cdot 10^{-3}$	$2{,}8 \cdot 10^{-4}$

[1)] Defect charcoal filter in the ventilation system in March and April

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Influence on radiation protection when increasing the power in nuclear reactors

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An increase of thermal power has been accomplished in most Swedish nuclear power plants during the years, with an acceptable radiological impact. However in the future more and larger increase in power is planned. An increase of the power in the order of 20-30 % has been discussed. The purpose of this paper is to describe and discuss SSIs view on how and to what extent the increase of power will have influence on radiation protection issues such as occupational exposure, effluence and discharges and waste management.

Regulatory control of the initial phases of decommissioning of the nuclear facilities at Risø.

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Abstract: Physical decommissioning of the nuclear facilities at Risø was launched on the 28th of October 2004, when the recombiner was disconnected from the primary system and lifted out of the smallest research reactor DR1. Prior to this date the regulatory authorities accomplished a multitude of preparatory tasks in order to implement the Danish legislation on decommissioning of the nuclear facilities at Risø, as well as the recommendations set out by the International Atomic Energy Agency. This contribution focuses on some of these tasks including a) aspects of the new Operational Limits and Conditions with emphasis on administrative structure and clearance criteria and b) the evaluation of the general decommissioning plan. It is concluded that: a) international recommendations provide a comprehensive basis for the national institutions involved in decommissioning, and b) frequent and early exchange of ideas in all aspects of decommissioning which require the nuclear regulatory authorities' approval may advance the process considerably.

Introduction

In September 2000, Risø National Laboratory's Board of Directors decided that the largest reactor, DR3, would not be restarted after an extended shutdown for inspection. At this point a phase-out of DR3 was anticipated and, as this was by far the largest facility; the decision to shut it down triggered the general motion to phase-out and decommission all of the remaining nuclear facilities at Risø. It was subsequently considered appropriate to segregate the decommissioning task from Risø National Laboratory into an independent institution with its own management.

Danish Decommissioning (DD) was established as an independent institution under the Ministry of Science, Technology and Innovation. On the 15th of September 2003, the responsibility for operation and decommissioning of the nuclear facilities, as well as continued waste treatment, was transferred from the Risø National Laboratory to DD as stipulated in Parliamentary Decision B48, by the 13th of March, 2003. Parliamentary Decision B48 also predetermined the decommissioning strategy: Immediate decommissioning, primarily dictated by the availability of experienced personnel, especially staff of the former operating organisation.

Revision of the Operational Limits and Conditions

One of the more important prerequisites to actual decommissioning was a thorough revision of the existing Operational Limits and Conditions in order to ensure that the decommissioning would be managed in an appropriate way. The new "Operational Limits and Conditions for DD" were issued by the nuclear regulatory authorities in October 2004. They include provisions on: Administrative structure, project planning and management,

detailed operation planning, quality assurance, characterization of radioisotope inventory, operational radiation protection, safety assessment, environmental impact assessment, documentation, and specific criteria for clearance.

Initial efforts were focused on: a) creating an adequate administrative structure with a clear-cut organisation and responsibilities within Danish Decommissioning, and b) generation of a general decommissioning plan with clear strategies and priorities. Other aspects were dealt with subsequently and e.g. the clearance criteria were included a while after the operational limits and conditions for DD went into force. These issues were all shaped in close agreement with IAEA Safety Standards with respect to both structure and essence, although further provisions were appended as exemplified below.

Administrative structure: In accordance with IAEA guidelines, the administrative structure within Danish Decommissioning was set up to ensure that the functions of safety, environmental protection and quality assurance would be independent from the units directly responsible for accomplishing the decommissioning activities. This structure was approved by the Danish Agency for Governmental Management. The safety organization can thus examine decommissioning projects independent from the project administration. However, the nuclear regulatory authorities furthermore ensured that the safety organization became independent from the financial administration. Moreover, in order to prevent unanticipated administrative barriers, it was ensured that employees may independently refer directly to the Director in matters fundamental to safety and security. Changes to this administrative structure are subject to the nuclear regulatory authorities' approval.

Clearance: Clearance criteria, which set out mass specific and surface specific clearance levels, are given in the Operational Limits and Conditions for DD. The criteria adhere to IAEA recommendations. However, in supplement, surface specific clearance levels are those recommended by the European Commission. For mass specific clearance levels for clearance of solid materials for disposal, reworking or reuse, the Operational Limits and Conditions for Danish Decommissioning refer directly to Table 2 in the IAEA Safety Guide No. RS-G-1.7, Application of the Concepts of Exclusion, Exemption and Clearance. For surface specific clearance levels for clearance of buildings for disposal, recycling or reuse the Operational Limits and Conditions for DD refer directly to European Commission, Radiation Protection 113, Recommended radiological protection criteria for the clearance of buildings and building rubble from the dismantling of nuclear installations, 2000. These values are also valid for the reuse of objects.

The nuclear regulatory authorities have set out additional requirements with reference to specific international standards e.g. for the competence of the laboratory characterizing, measuring, handling and sorting waste. It is thus required that the clearance laboratory by the end of 2006, is accredited according to the DS/EN ISO17025 standard on general requirements for the competence of testing and calibration laboratories. By the use of external laboratories these must also be accredited according to the above standard.

Furthermore, the nuclear regulatory authorities have specified the concept of good practice in the section on averaging of activities. For masses of up to 1000kg with average activity concentrations below the clearance levels, but where activity concentrations which exceed the clearance levels have been identified in a segment of the entire volume/mass, this segment must be removed if reasonably achievable.

Evaluation of the general decommissioning plan

As the general decommissioning plan defines the strategy by which the physical decommissioning is carried out, it has bearings on the detailed decommissioning plans for each of the nuclear facilities. Setting up a general decommissioning plan is thus a challenging procedure, in which the operator must take the technicalities of each individual facility into account in a general way and fix an order of priority for a multitude of widely different tasks of e.g. technical or administrative character. In the present case correspondence regarding the general decommissioning plan developed into an extensive iterative process. As a consequence the Operational Limits and Conditions for DD states directly that the decommissioning plans should be based on the guide given in the IAEA Safety Guide No. WS-G-2.1, Decommissioning of Nuclear Power Plants and Research Reactors.

The most significant provisions and suggestions raised by the nuclear regulatory authorities were the following: a) In order to facilitate a systematic approach by the nuclear regulatory authority, adhere closely to the IAEA guideline for the general decommissioning plan and use a logical and recurring structure in each chapter, b) refer to national or international recommendations or experience whenever possible, c) define the decommissioning strategy clearly; a unifying principle should transform into operational procedures and instructions without problems, d) avoid listings of potential solutions to problems without stating clearly what information is needed in order to make the optimum choice, and e) always state clearly the information behind any selection of approach, method or technique.

Conclusions

In conclusion; international recommendations provide a fundamental and comprehensive guideline to the national institutions and organizations involved in decommissioning. However, beyond the given essential regulatory provisions some additions may be appropriate. With respect to the interaction between the operator and the authorities; our experience so far stress the importance of frequent and early exchange of ideas in all aspects of decommissioning which require the nuclear regulatory authorities' approval.

Effluent release options from nuclear installations

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NEA Report

Effluent Release Options from Nuclear Installations: Technical Background and Regulatory Aspects, OECD (2003) ISBN 92-64-02146-9 (available also in web):

- To provide basic factual information on different options for managing and regulating radioactive effluent releases from nuclear installations during normal operation.
- To contribute to national and international discussions in this area.
- To assist the development of national approaches to effluent release management

Setting the scene – the report provides information on

- International and intergovernmental agreements and declarations,
  - Convention on Nuclear Safety, Joint Convention, Euratom Treaty, OSPAR Convention, Sintra Statement, Rio Conference,
- Activities of international and intergovernmental organisations,
  - International Atomic Energy Agency (IAEA), United Nations Scientific Committee on the Effects of Atomic Radiation (UNSCEAR), European Commission (EC), World Health Organisation (WHO), Food and Agriculture Organisation (FAO),
- National policies,
  - Belgium, Finland, France, Germany, Japan, Spain, Sweden, United Kingdom, Unites States.

- **Current effluent releases from nuclear installations**

See for example **time trends in normalised releases of radionuclides for PWR reactors**, based on data published in the latest UNSCEAR report [2],

As low as reasonably achievable (ALARA)

- System of radiological protection, as described in ICRP Publication 60, is based on the three basic principles:
  - justification of a practice,
  - optimisation of protection and individual dose, and
  - risk limits.
- Optimisation of protection for practices, introducing the term ALARA, focuses on individual doses and refers to risks assessed using the dose/risk relationship recommended by the ICRP.
- ALARA has proven to be an **effective tool for managing human risks** after low dose exposures taking into account individual doses, the number of exposed individuals and the likelihood that an exposure situation will occur.

Best available techniques (BAT)

- Concept used in the *OSPAR Convention for the Protection of the Marine Environment of the North-East Atlantic* for all types of industrial installations including nuclear installations.
- Set of common rules in the *Integrated Pollution Prevention and Control (IPPC) Directive of 1996*, to minimise pollution from various point sources throughout the European Union.
  "best available techniques" *shall mean the most effective and advanced stage in the development of activities and their methods of operation which indicate the practical suitability of particular techniques for providing in principle the basis for emission limit values designed to prevent and, where that is not practicable, generally to reduce emissions and the impact on the environment as a whole.*
  - ***"techniques"*** *shall include both the technology used and the way in which the installation is designed, built, maintained, operated and decommissioned;*
  - ***"available"*** *techniques shall mean those developed on a scale which allows implementation in the relevant industrial sector, under economically and technically viable conditions, taking into consideration the costs and advantages, whether or not the techniques are used or produced inside the Member state in question, as long as they are reasonably accessible to the operator;*
  - ***"best"*** *shall mean most effective in achieving a high general level of protection of the environment as a whole.*
- The BAT approach ensures that the cost of applying techniques is not excessive in relation to the environmental protection they provide. It follows that the more environmental damage BAT can prevent the more the operator can justify spending before the costs are considered excessive.
- Determining BAT involve identifying options, assessing environmental effects, considering economics, and applying the principles of precaution and prevention.
- BAT should help to choose the best option available to achieve a high level of protection of the environment taken as a whole.

ALARA and BAT

- ALARA and BAT are both optimisation approaches complementing each other with the aim of limiting doses to humans, possible effects on non-human species, and radioactive effluent releases.
- ALARA and BAT are both moving targets, since developing societal values and advancing techniques may change what is currently regarded as "reasonably achievable" and "best available".

Use of the concepts ALARA and BAT in effluent release management

Technical, societal and policy factors influencing decisions on effluent release options

Decision-aiding strategies for effluent release options

Summary

- BAT and ALARA share much common ground, the factors which influence BAT are different and wider than ALARA's health protection focus. Further practical guidance is needed to illustrate BAT approach in different cases for operating or designed NPPs. National cases are existing and they could be discussed e.g. in a dedicated NEA Workshop.

Inspections on individual monitoring of internal contamination using whole-body counters

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Abstract: In Sweden the nuclear facilities use whole-body counters to monitor internal contamination of the workers to assess an actual intake of radioactive substances. From the determined intake the committed effective dose can be calculated. The Swedish Radiation Protection Authority (SSI) has issued occupational radiation protection regulations for nuclear facilities, which include general demands in the area of monitoring and calculation of internal doses to workers. SSI has performed inspections to check that the facilities fulfil the regulatory demands concerning whole-body monitoring. The inspections, which took part in 2004 and 2005, consisted of checking manuals, equipment and relevant quality documentation. Moreover, the facilities have been requested to present their result from the last inter-calibration work. In this presentation SSI will describe the procedure, the different parts of the activities that have been checked and present the conclusions from the inspection work.

Background

In the beginning of 2004 Swedish Radiation Protection Authority (SSI) decided to perform topical inspections in the area of internal contamination monitoring at the Swedish nuclear facilities. The main objective was to find out whether the facilities fulfil the basic requirement in the national regulations. It was also intended to be a part of the Swedish system for legal approval of dosimetric services in accordance with the EC Council Directive 96/29/Euratom.

Swedish nuclear facilities

In Sweden there are four sites with a total of 12 nuclear power reactors, one site with a research reactors and other nuclear activities such as radioactive waste facilities, and finally one nuclear fuel factory. All these sites (six altogether) have laboratories that use whole-body counters to monitor personnel for internal contamination.

Legal Framework

The legal framework in the area of occupational radiation protection at nuclear facilities consists of several regulations issued by SSI. In the activity of monitoring of internal contamination the relevant legal conditions are specified in *Regulations on Radiation Protection of People Exposed to Ionizing Radiation at Nuclear Plants (SSI FS 2000:10)*. Paragraph 23 in this regulation sets out the requirements on monitoring of internal contamination with whole-body counters and was the base for the inspection performed. The paragraph reads:

> § 23 Whole body counting shall be performed according to a documented procedure that is approved by the SSI. The documentation shall show
>
> 1. the measurement equipment and routines for its use,

2. the competence of the persons performing the measurements,
3. methods used for calculating the intake and the committed effective doses,
4. routines for evaluating the obtained results and
5. routines for calibrations and checks of the equipment.

Inspection method

All inspections took place during October and November 2004. To be properly prepared for the inspections SSI asked the licensees for documents in advance that briefly described the activities connected with whole-body measurements and their view on how they fulfil the requirements above. SSI also prepared an inspection questionnaire that consisted of 27 questions. The questions were grouped in different areas:

- Organization: In this area SSI checked the existence of relevant economical and personal resources, competence of personnel as well as educational plans.
- Quality systems: The objective in this area was to make sure the activity of whole body monitoring was covered in the overall quality system at the company.
- Quality documents: In this area SSI checked for documents describing relevant routines in monitoring, documentation of results, reporting of results, investigation levels, routines for handling an actual intake and so on.
- Technical equipments: SSI checked the equipment, the existence of relevant technical manuals and calculation methods.
- Calibration: In this area SSI checked for calibration routines and periodicity and routines for documentation of performed calibrations.
- Quality auditing: In this area SSI checked that whole-body measurement activities are taken care of in relevant company auditing systems.

The questionnaire was sent to the licensees in advance, to make it possible to prepare relevant and necessary documentations.

Inspection performance

Every inspection was performed during one day. Studying the prepared documentation and equipment was the main work. After every single question in the inspection questionnaire SSI made one of the following conclusions: Fulfilled, fulfilled (with remarks) or not fulfilled. In the end of each inspection SSI presented its findings to the representatives of the company and discussed and agreed upon a relevant time limit for the company to make corrective actions. A brief inspection reports, including the findings, were sent to each of the licensees within a week.

Results from the inspections

The topical inspections at the Swedish nuclear facilities on monitoring of internal contamination using whole-body counters resulted in some findings, which have been summarized according to every area of interest (table 1). As can be seen the number of findings were quite few and most of them were of the kind that they could be corrected in a fairly easy manner.

Table 1. Findings during the inspections

Area of inspection	Number of laboratories with findings	Total number of findings
Organization	2	6
Quality system	1	1
Quality documents	3	8
Technical equipment	1	1
Calibration	3	7
Auditing system	1	1

Actions were taken to deal with the findings in a few months and in the end of the spring 2005 all the laboratories received a confirmation from SSI that they do perform their whole-body measurement activities in according with the regulations from SSI.

Conclusions

SSI has concluded that the six nuclear facilities in Sweden, using whole-body counters to monitor internal contamination, fulfill the specific conditions that are set out in the Swedish regulation. They are now approved as dosimetric services in accordance with EC Council directive 96/29/Euratom.

Session VI: Non-ionising radiation

Non-ionising radiation protection – an introduction

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Abstract: This paper presents briefly non-ionising radiation i.e. optic and radiofrequency radiation and its biological effects. There are also listed reasonably well known exposure limits.

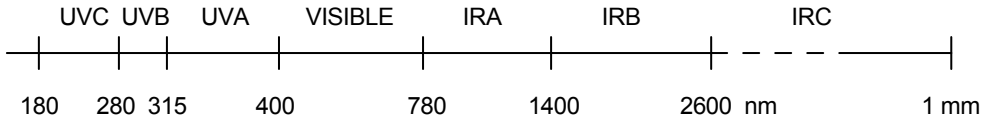
Optical radiation

Natural optical radiation is given by the sun and is a necessity for all life upon earth. Humans, animals and plants are also on the whole adapted to the natural optical radiation and benefit from it. The margins are however small. Many different kinds of man-made sources are capable to give radiation levels which are several orders of magnitude larger than the natural radiation and careless exposure to such sources might cause damages. The organs affected are skin or eye.

Spectrum

By definition optical radiation is electromagnetic radiation of wavelengths between 100 nm and 1 mm, (10^{-7} – 10^{-3} m). However the atmosphere completely absorbs radiation with the shortest wave-lengths, so from a radiation protection officers point of view the story starts at about 180 nm.

The optical spectrum is divided in three main parts, ultraviolet (UV), visible and infrared (IR), each with sub-intervals generally with borders and designations as shown in figure 1. As shall be shown there are biological reasons given by the human eye depending on the depths of penetration why the borders are as they are with only one exception.



The optical spectrum

To the left of the spectrum (shorter wavelengths) there are x-rays and gamma-radiation, to the right there are the different bands of radiofrequency (microwaves, TV-signals and broadcasting)

Absorption in the eye

UVC and UVB is strongly absorbed already in the epithelium i.e. the farthest out cell layer of the cornea (figure 2). The border at 280 nm is not biological but is there because 280 nm is the shortest wavelength in the natural radiation from the sun hitting the earth's surface.

Outside the atmosphere strong UVC radiation is present but it is all absorbed, mainly in the ozone-layer.

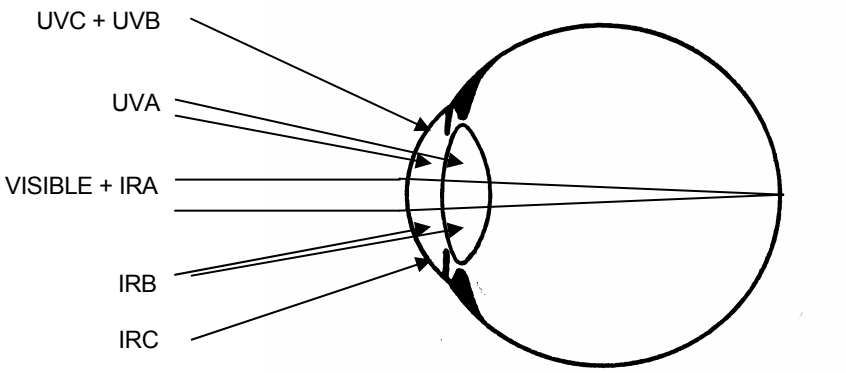
UVA penetrates the cornea and is absorbed in the aqueous and the lens.

Visible radiation is transmitted very easily to the retina. The commonly used limits 400 and 780 nm are not sharp. You can probably see a strong 390 nm source. In many standards, for example dealing with safety goggles, the interval 380 to 780 nm is referred to as visible. Most people are also able to see something of the radiation from a 820 nm GaAlAs laser diode.

IRA is also transmitted to the retina but gives no visual sensations. With increasing wavelength the absorption in the vitreous increases, and 1400 nm wavelength is the longest possible to reach the retina.

IRB is absorbed similarly as UVA i.e. in the aqueous and the lens. 2600 nm is the longest wavelength that can pass the cornea.

IRC is absorbed in the cornea.



Absorption in the eye

Skin

In the skin optical radiation penetrates to different depths depending on the wavelength. Generally the longer wavelength, the deeper penetration. However this does not mean more than fractions of millimetres up to a few millimetres. Depending on the wavelength and thus on the depth of penetration different hazards mechanism may occur. This depends on what kind of cells are hit and affected.

It is a well known fact that prolonged and intense exposure to the ultraviolet radiation may result in skin-cancer. The solar radiation in Scandinavia is a serious radiation protection problem seen to public health. Also the use of solaria can not be ruled out in this respect. There is a recent common statement among the Nordic countries that does not recommend the use of solaria.

There are three groups of skin-cancer: basal cell carcinoma, squamous cell carcinoma and melanoma. The two first types are relatively easy to treat but the melanomas may be mortal.

The two first types can obviously be related to exposure to UV, in the first place UVB from the sun. When it comes to melanoma, prolonged exposure to ultraviolet is obviously one promoter but the action spectrum is not well known.

Erythema

Erythema is an acute effect which arises a few hours after exposure to UV and remains a few days. Large doses - say 10 times above the exposure limits - may cause painful blisters. As is shown by the action spectrum (Table 1) there is a strong dependence with wavelength in the ultraviolet spectrum.

Hazards to the eye

Ultraviolet radiation

A cell dies if it receives a large enough dose of UVC or UVB. This is often used in laboratories and operating rooms where one wants a sterile environment. This also causes troubles in the cornea where such radiation is strongly absorbed. The dead cells are swept away by the eyelid, leaving bare nerve-ends which are extremely sensitive. This painful state, known as photokeratitis or snow blindness, begins a few hours after an overexposure. The injury will heal within a 24 hours period of time with no remaining defects.

UVA is not as aggressive as the shorter wavelengths of ultraviolet. Being absorbed partly in the lens the possibility that UVA causes cataract can not be ruled out or at least takes part in that process. As being a long-term effect it is hard to verify a relationship.

Visible radiation

1. Retinal burns. Visible radiation is transmitted to and focused on the retina. An almost parallel laser beam or a small bright source such as a welding arc (bright sources generally are small) focused on the retina produces heat where the radiation is absorbed. Too much absorption will burn the receptors which are destroyed and never will come back. Here is a threshold mechanism. Either a receptor is overheated and destroyed or it is not and then nothing happens. If overheated the process goes very rapidly - a few seconds down to microseconds.

An exposure a couple of magnitudes larger than the threshold is likely to puncture a blood-vessel in the retina and blood is floating into the vitreous. Still a couple of magnitudes larger an exposure will produce a high pressure steam blister in the retina which may completely destroy the eye. The only sources known that are capable to produce damages of the two latter types are high energy pulsed lasers.

2. Photochemical effects. Also a non-thermal hazard threatens the retina. The mechanisms are not fully understood but could be described as if the photochemical processes in the receptors, the cones, giving visual sensations go too fast in too bright light. The cones are not dead but may be sleeping for weeks.

Infrared radiation

Also IRA is transmitted and focused on the retina though there is an increasing absorption in the vitreous with longer wavelengths. The hazards from thermal effects as described under visible radiation remain but there is no reason to fear photochemical effects.

In fact the reported cases of retinal burns are dominated by accidents with NdYAG-lasers with wavelength within IRA (wavelength 1060 nm). The invisibility of the beam takes for sure part in many such accidents.

As early as in the end of the seventeenth century it was observed that elderly glass-blowers often got cataracts, but not until 1984 a Swedish (!) study convincingly showed that there is a relation between prolonged industrial exposure to infrared and cataract. It is reasonable to believe that IRB plays an important part in this process since the radiation absorbed by the lens to an overwhelming extent is IRB.

The study was made in steel industry as well as in glass works. The conclusion is that at an age of 70 years the probability to get a cataract is 10 times greater for those who have been exposed to infrared regularly in their work compared to non-exposed workers. With infrared exposure is meant that the temperature is such that the material glows (i.e. 800 K or more), the geometry is such that a sensation of heat is obvious in the skin and the duration of exposure have been the major part of the working day for many years.

IRC and its strong absorption in the cornea implies heating of the cornea with possible corneal burns or drying as a result. In my world such hazards only exists if you are taken by surprise by a powerful source, for instance a long-wave laser beam. The cornea is extremely sensitive to pain and will immediately tell if something is wrong. Therefore it is not likely that people stand calmly in front of a steel furnace until they get their corneas burnt.

Exposure limits

In ultraviolet, visible and infrared A the biological effects are acute and thus easy to study in laboratory experiments. For these wavelengths there are generally accepted exposure limits (Tables 1 and 2). For longer wavelengths the effects are long-term and here is a lack of knowledge which long-term exposure limits should be reasonable.

Radiofrequency radiation

When it comes to radio-frequency, commonly referred to as electromagnetic fields (EMF) in spite of the fact that x-rays and optical radiation are also electromagnetic, there are a number of biological effects at over-exposures, not necessarily harmful but observable and unwanted.

Electric and magnetic fields create electric currents in all conductors including human tissues. These currents may interfere with the electric signals that normally shall take place in the nervous system or produce heat.

Going over to the radiofrequency part of the electromagnetic spectrum the concept frequency is more commonly used rather then the wavelength.

The relationship is simple: $f \cdot \lambda = c$

where f is the frequency, λ is the wavelength and c is the speed of light.

Biological effects

At exposure to low frequency radiation (long wavelengths, kilometres or more) the human body (up to 2 metres) is not a well adjusted antenna. Therefore very little energy is absorbed in the human body giving no problems with heat sensations. The limiting factor is the induced current in the body. At higher frequencies the human body or the organs connect much more energy and heat is produced.

0 - 1 Hz: Overexposure to magnetic flux density (T) due to static fields or for time varying fields up to 1 Hz current density (A/m^2) in the body may affect the heart or the blood circulation.

1 Hz - 10 MHz: The critical quantity is the current density (A/m^2) in the body that may affect the central nervous system.

100 kHz - 10 GHz: A too high value of SAR (specific absorption rate, W/kg) may heat the whole body or a part of it. In the frequency region 100 kHz - 10 MHz also effects on the central nervous system may be of concern. For higher frequencies SAR is the limiting quantity.

10 GHz - 300 GHz: This radiation heats the skin or tissues near the skin. The depth of penetration is small. The limiting quantity is SAR.

Pulse: A short (microseconds) large energy pulse in the frequency region 0,3 GHz - 10 GHz may produce sound sensations. This is known as "microwave hearing" as clicking or popping sounds. Repeated or prolonged exposure may be stressful and potentially harmful.

Exposure limits

One of the most developed mapping of biological effects by EMF is made by the International Commission on Non-Ionising Radiation Protection (ICNIRP), a sister organisation to ICRP. ICNIRP has gone through hundreds of scientific reports and made an evaluation of them. ICNIRP has also recommended exposure limits for the general public and respectively for workers in practices where EMF is involved. ICNIRPs method was to determine what exposure levels in different frequency intervals give observable effects in humans. These levels were then divided by 10 for workers and 50 for the general public and thus exposure limits were obtained which at present are considered to be acceptable. The difference by about a factor of five is due to the fact that workers are supposed to be adults and healthy while the general public includes children and old people.

The exposure limits consist of basic restrictions, which are derived from phenomena inside the human body. These are generally not measurable outside a laboratory. In stead there are sets of physical quantities, measurable in the environment, which if not exceeded guarantee that the basic restrictions are fulfilled.

The recommendations settled by ICNIRP for the general public are shown in Tables 3 and 4. These recommendations are also adopted by EC and as in many other countries appear in general advice from the Swedish Radiation Protection Authority.

Table 1 Exposure limits (EL) and efficiency factors (Sλ) for ultraviolet radiation in an 8-hours period of time

λ (nm)	EL (J/m²)	Sλ	λ (nm)	EL (J/m²)	Sλ
180	2500	0,012	310	2000	0,015
190	1600	0,019	313*	5000	0,0060
200	1000	0,03	315	10000	0,0030
205	590	0,051	320	29000	0,0010
210	400	0,075	325	60000	0,00050
215	320	0,095	330	73000	0,00041
220	250	0,12	335	88000	0,00034
225	200	0,15	340	110000	0,00028
230	160	0,19	345	130000	0,00024
235	130	0,24	350	150000	0,00020
240	100	0,30	355	190000	0,00016
245	83	0,36	360	230000	0,00013
250	70	0,43	365*	270000	0,00011
254*	60	0,50	370	320000	0,000093
255	58	0,52	375	390000	0,000077
260	46	0,65	380	470000	0,000064
265	37	0,81	385	570000	0,000053
270	30	1,00	390	680000	0,000044
275	31	0,96	395	830000	0,000036
280*	34	0,88	400	1000000	0,000030
285	39	0,77			
290	47	0,64			
295	56	0,54			
297*	65	0,46			
300	100	0,30	* Wavelengths by		
303*	250	0,12	a mercury lamp		
305	500	0,060			

If a source gives several wavelengths at the same time the following is valid:

$$EL = \Sigma\ t \cdot (E\lambda \cdot S\lambda \cdot \Delta\lambda) = 30\ J/m^2$$

where t is the (maximum) exposure duration in seconds and Eλ is the spectral irradiance (W/m^2nm) at the wavelength λ.

Table 2 Efficiency factors for blue-light hazards (Bλ) and retinal burn hazards (Rλ)

λ	Bλ	Rλ
400	0,10	1,0
405	0,20	2,0
410	0,40	4,0
415	0,80	8,0
420	0,90	9,0
425	0,95	9,5
430	0,98	9,8
435	1,0	10
440	1,0	10
445	0,97	9,7
450	0,94	9,4
455	0,90	9,0
460	0,80	8,0
465	0,70	7,0
470	0,62	6,2
475	0,55	5,5
480	0,45	4,5
485	0,40	4,0
490	0,22	2,2
495	0,16	1,6
500-600	$10^{(450 - \lambda)/50}$	1,0
600-700	0,001	1,0
700-1049	-	$10^{(700 - \lambda)/505}$
1050-1400	-	0,2

The exposure limits are given by the following expressions:

Blue-light hazard:

$t \cdot \Sigma(L\lambda \cdot B\lambda \cdot \Delta\lambda) \leq 10^6$ J/m²sr
if $t \leq 10^4$ s,

and $\Sigma(L\lambda \cdot B\lambda \cdot \Delta\lambda) \leq 100$ W/m²sr
if $t > 10^4$ s.

Burn hazard:

$\Sigma(L\lambda \cdot R\lambda \cdot \Delta\lambda) \leq 5 \cdot 10^4/(\alpha \cdot t^{1/4})$ $t \leq 10$ seconds

t = duration of exposure (seconds)
Lλ = spectral radiance of the source (W/cm²sr)
α = largest plane visual angle to the source (radians)

Table 3 Basic restrictions for exposure to EMF

Frequency range	Magnetic flux density (mT)	Current density (RMS-value) (mA/m^2)	SAR (Mean over the body) (W/kg)	Local SAR (head and trunk) (W/kg)	Local SAR (Arms and legs) (W/kg)	Power density (W/m^2)
0 Hz	40	-	-	-	-	-
> 0 Hz - 1 Hz	-	8	-	-	-	-
1 Hz - 4 Hz	-	8/f	-	-	-	-
4 Hz - 1 kHz	-	2	-	-	-	-
1 kHz - 100 kHz	-	f/500	-	-	-	-
100 kHz - 10 MHz	-	f/500	0,08	2	4	-
10 MHz - 10 GHz	-	-	0,08	2	4	-
10 GHz - 300 GHz	-	-	-	-	-	10

f is the frequency (Hz).

The current density should be evaluated as the mean value over an area of 1 cm^2 perpendicular to the current.

The SAR-values (specific absorption rate) regard the mean value in a 6 minutes period of time.

Local SAR should be evaluated as the mean value in a mass of 10 g.

Table 4 Recommended exposure to EMF limits for the general public

Frequency range	Electric field strength (E) (V/m)	Magnetic field strength (H) (A/m)	Magnetic flux density (B) (μT)	Power density S_{eq} (W/m^2)
0 Hz - 1 Hz	-	$3{,}2 \cdot 10^4$	$4 \cdot 10^4$	-
> 1 Hz - 8 Hz	10000	$3{,}2 \cdot 10^4 / f^2$	$4 \cdot 10^4 / f^2$	-
8 Hz - 25 Hz	10000	4000/f	5000/f	-
25 Hz - 800 Hz	$2{,}5 \cdot 10^5 / f$	4000/f	5000/f	-
800 Hz – 3 kHz	$2{,}5 \cdot 10^5 / f$	5	6,25	-
3 kHz - 150 kHz	87	5	6,25	-
150 kHz - 1 MHz	87	$7{,}3 \cdot 10^5 / f$	$9{,}2 \cdot 10^5 / f$	-
1 MHz - 10 MHz	$8{,}7 \cdot 10^4 / f^{1/2}$	$7{,}3 \cdot 10^5 / f$	$9{,}2 \cdot 10^5 / f$	-
10 MHz - 400 MHz*	28	0,073	0,092	2
400 MHz - 2 GHz	$\frac{1{,}375 \cdot f^{1/2}}{1000}$	$\frac{0{,}0037 \cdot f^{1/2}}{1000}$	$\frac{0{,}0046 \cdot f^{1/2}}{1000}$	$f/(2 \cdot 10^8)$
2 GHz - 300 GHz	61	0,16	0,20	10

f is the frequency in Hz.

In the frequency range between 100 kHz and 10 GHz, S_{eq}, E^2, H^2 and B^2 should be evaluated as the mean value during 6 minutes.

At a frequency greater than 10 GHz, S_{eq}, E^2, H^2 and B^2 should be evaluated as the mean value during $68/(f \cdot 10^{-9})^{1{,}05}$ minutes, where f is expressed in Hz.

Measurements of radiofrequency electromagnetic fields in different environments

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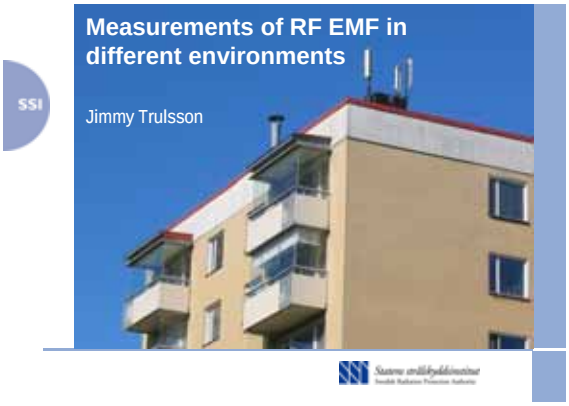
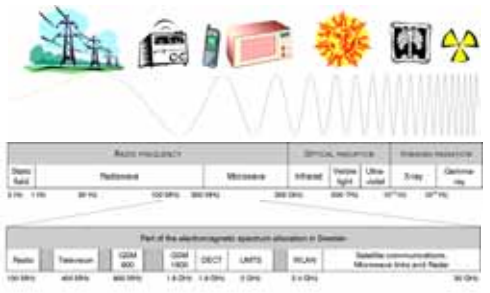
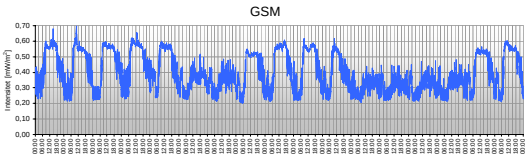
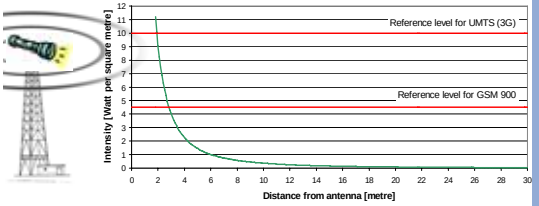
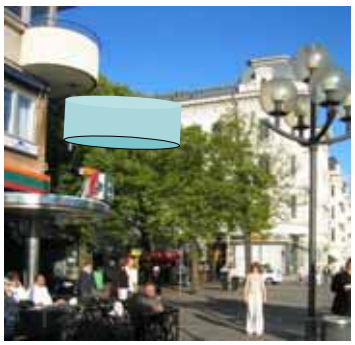


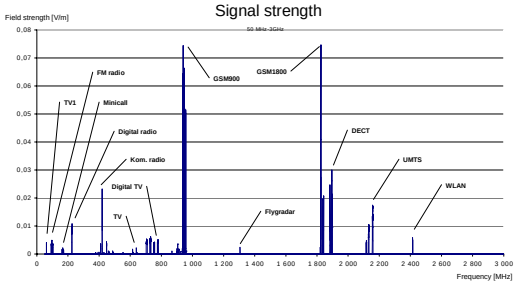


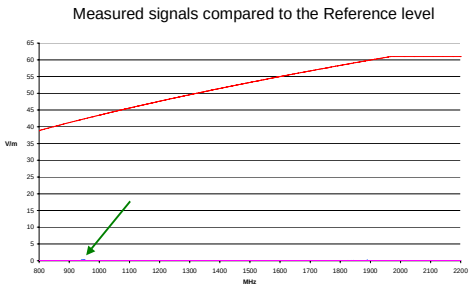
Base station for GSM900	40 Watt
Base station for 3G	20 Watt
Mobile phone for GSM900	0,25 Watt
Mobile phone for 3G	0,125 Watt
Wireless computer network (WLAN)	0,1 Watt
Cordless phone (DECT)	0,01 Watt
TV broadcasting	40 000 Watt max 1 000 000 Watt











Percent of Reference level

Radio	0,0099
TV	0,0094
Mobil	0,9953
DECT	0
Övrigt	0,0045
Total	*1,02%*

GSM	0,098 %
3G	0,043 %
TV+radio	0,205 %

3G	0,00007 %
TV+radio	0,003 %

GSM900-Base stations	0,091 %
GSM1800-Base stations	0,026 %
3G-Base stations	0,460 %

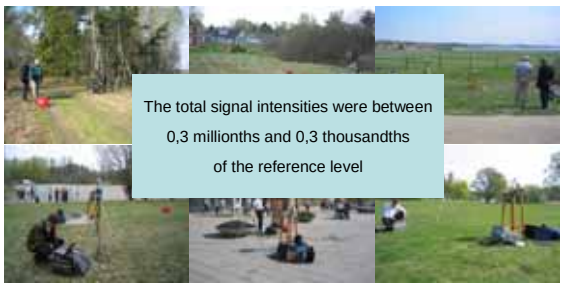
On the platform

GSM900-Base stations	0,0130 %
GSM900-Phones	0,0022 %

In the underground train

GSM900-Base stations	0,0016 %
GSM900-Phones	max 0,22 %

Measurements at ten sites in outdoor environments
(Ekerö and Solna)



The total signal intensities were between
0,3 millionths and 0,3 thousandths
of the reference level

Conference room WLAN-office

Conclusions and comments

The exposure to general public from base stations is far below the reference levels.

Your own mobile phone causes much higher exposure than the surrounding base stations do.

The use of "hands-free" can reduce your exposure by 99 %.

An effective way to reduce your exposure is therefore to use a "hands-free" device.

Session VII: Medical radiation

Medical applications of luminescence dosimetry using fibre-coupled Al_2O_3:C crystals

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Abstract: We have developed a new in-vivo real-time dosimetry system that uses radioluminescence and optically stimulated luminescence from 2 mg crystals of Al_2O_3:C attached to 15 m long optical fibre cables. The prime application of the system has hitherto been in-vivo dose verification in intensity-modulated radiation therapy. The system has also shown its potential in diagnostic radiology when applied to mammography. This paper outlines the capabilities of the system in medical applications and its potential in general radiation protection is discussed.

Introduction

We have developed a new luminescence dosimetry system based on small detector probes. Each probe consists of a 2 mg crystal of carbon-doped aluminum oxide (Al_2O_3:C) attached to a 15 m long optical fibre cable (PMMA). The prime application of the system has so far been in-vivo dose verification in intensity-modulated radiation therapy (IMRT), but the system has also shown its potential in diagnostic radiology when applied to mammography (Aznar et al., 2004).

The system provides independent measurements of absorbed dose-rate and dose. The dose-rate measurements are based on radioluminescence (RL) generated in the crystal during the radiation treatment. This signal is similar to the one that is generated in a scintillator. Measurements of the integrated dose (e.g. from a single treatment fraction) is obtained from optically stimulated luminescence (OSL). The readout of this signal is made after the treatment using a green laser as stimulation source.

The key features of the system are as follows: (1) High precision and accuracy, (2) small detector size, (3) real-time readout, (4) large dynamic dose range, (5) two signals (one for dose rate and another for the integrated dose), (6) no stem effect with pulsed linear accelerator beams, (7) passive dosimeter probe without, for example, electrical wires (i.e. the probe is suitable for in-vivo dosimetry).

The system has been developed jointly with partners in the USA (Landauer Inc. and Oklahoma State University) and the clinical tests have mainly been carried out at the university hospitals in Copenhagen and Malmö.

System

The dosimeter system is outlined in Figure 1. A photomultiplier tube (Perkin Elmer, MP982) and a 395-440 nm band pass filter is used for luminescence detection, and optical stimulation is carried out with a 532 nm 20 mW laser. The synchronization signal from the linear accelerator is used to separate the RL signal from the unwanted stem signal (i.e. short pulses of fluorescence and Cerenkov light generated in the fibre cable with

each linac beam pulse). A simple algorithm is used to compensate for RL-sensitivity changes. A more comprehensive description of the RL/OSL measurement system and how it is calibrated can be found elsewhere (Aznar et al., 2004 and Andersen et al., 2005).

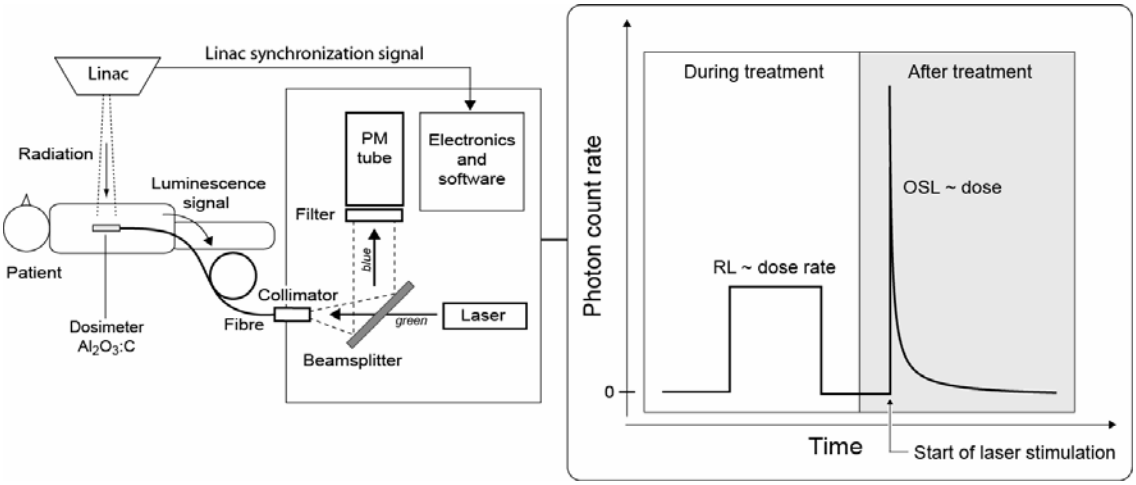

Figure 1. Outline of the luminescence dosimetry system.

Radiation therapy tests

Figure 2 shows a depth-dose curve obtained from the RL signal. The good agreement between the RL results and the reference Si-diode suggests that the system has a flat energy response for the relevant spectra for 6 MV photons.

Figure 3 shows a comparison between dose-rate measurements obtained using the RL-signal and a reference Farmer type ionization chamber (FC65-G Scanditronix Wellhöfer). Data were acquired at a rate of about 10 readings pr. second. As can be seen from the measurements, the dose rate provided by the linac (a Varian 2100 accelerator) is not stable during the first 5 - 10 seconds. There is, however, very good agreement between the results obtained with the luminescence system and the ionization chamber. This goes for both dose and dose-rate measurements. The luminescence system time-resolves the dose-delivery almost perfectly.

We have conducted measurements for a range of IMRT treatments using both the sliding window technique (i.e. dynamic MLC) and so-called step-and-shoot (Aznar et al., 2004 and Andersen et al., 2005). Most of the measurements have been carried out in solid-water phantoms, but a few have been with real patients. Using the RL-based dose rate measurements, we can precisely monitor the dose delivery for each treatment field. Comparison of results obtained when repeating the same treatments three or four times shows that we can determine the individual field doses with a precision of about 0.3%. Generally, the luminescence results are in very good agreement with the reference doses from the treatment planning system (the deviations are less than 2%) .

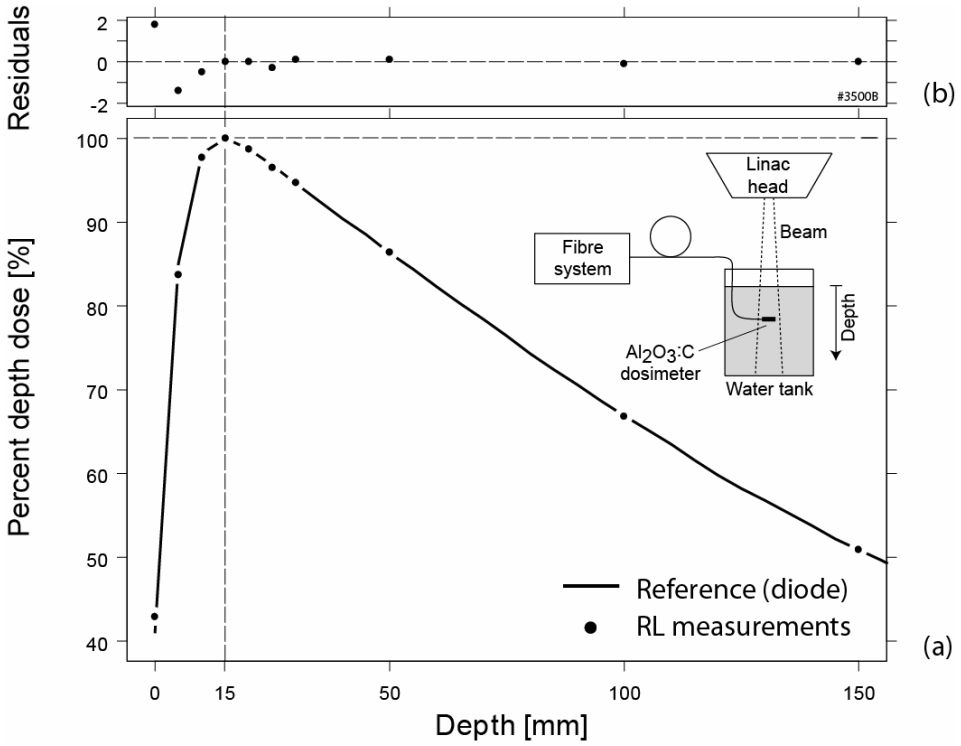

Figure 2. (a) Depth-dose curve for 6 MV photons, 100 cm SSD and 10 x 10 cm^2 field size. The results for both the RL and the diode measurements have been normalized to dmax at 15 mm. **(b)** shows the deviation between the RL measurements and the reference diode.

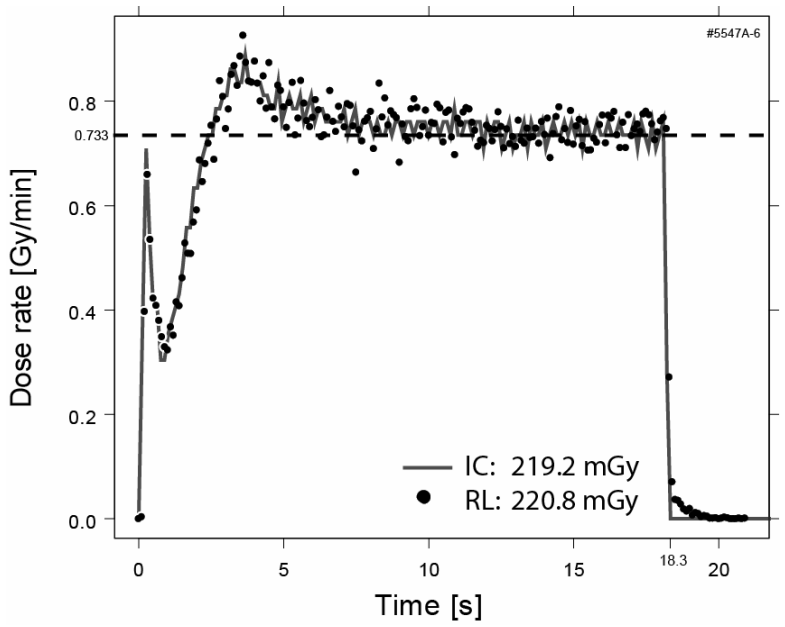

Figure 3. Measurement of the 6 MV dose delivery for a Varian 2100 linear accelerator using the RL technique (points) and a reference ionization chamber (IC; solid curve). The accelerator was set to give 220 mGy at the location of the luminescence probe with a dose rate of 733 mGy/min (30 MU at 100 MU/min). The transient feature within the first 5-10 seconds of the shot is a special characteristic for this particular accelerator. Integrating the dose-rate measurements gives the doses noted in the figure.

Other applications

The luminescence dosimetry system has a large dynamic dose range, and it has been used in both the Gy-range for radiation therapy and in the mGy-range for diagnostic radiology (Aznar et al., 2004). The RL-signal may only be useful for high dose rates, but the OSL allows for long integration times and can even be used to measure environmental doses. We would like to point out that PMMA fibre cables are very inexpensive and that we have carried out luminescence measurements using a 100 m long fibre cable. Hence, the system can be used for remote monitoring in places not normally accessible with larger instruments. Klein et al. (2005) have developed a special fibre-coupled Al_2O_3:C probe and a portable reader for monitoring of DOE nuclear waste sites. A minimum detectable dose of 15 μGy is possible after a 6 day exposure.

Conclusion

A new luminescence based dosimetry system has been developed. As outlined in this document, the system has excellent properties for medical applications such as in vivo dosimetry during radiation therapy. However, other applications within the field of remote sensing and radiation protection can be foreseen. Additional information about the system can be found in the listed references and at the web-site: http://www.risoe.dk/nuk/OSL-medical.htm.

Acknowledgements

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Diagnostic reference levels and measured administered activity to patients in Sweden

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Abstract: Since the 1960:s all hospitals in Sweden have to report the amount of activity used in nuclear medicine for all isotopes and all types of examinations. Diagnostic reference levels were introduced in 2002 for a number of specified examinations. All hospitals (34) using radioactive substances for in-vivo diagnostics had to measure the activity administered to 20 adult patients for the specified examinations. The compilation of data and conclusions drawn from the results will be presented as well as ideas for future modifications of the reference levels.

Introduction

Diagnostic reference levels were introduced in Sweden in 2002 for a number of specified examinations. All hospitals using radioactive substances for in-vivo diagnostics had to measure the activity administered to 20 adult patients for 19 specified examinations. Results and conclusions drawn from the compilation are presented as well as ideas for future modifications of the reference levels.

In 2003 approximately 103 000 examinations and 3300 therapies were performed in Sweden.

Tc-99m accounted for 96 % of all activity used. The total dose to the population from diagnostic procedures was 452 manSv.

Determination of Diagnostic Reference Levels

Diagnostic reference levels (DRL) were introduced with the aim to identify hospitals with procedures where too high activity is used with an unnecessary patient doses as a result. If the mean of the measured administered activities exceeds the diagnostic reference level the hospital have to investigate why and if possible take actions to reduce the administered activity.

The regulations about diagnostic reference levels in nuclear medicine states:

- DRLs for 19 examinations.
- Data for adults only.
- Determination of administered activity to at least 20 patients if more than 30 examinations per year are performed.
- If the average of the measured activity exceeds DRL corrective measures have to be done.
- Guidance on what to document.
- Demands for corrective procedures if values are exceeded.
- Measurements to be done at least every third year.

Since the 1960:s all hospitals in Sweden have to report the amounts of activity used in nuclear medicine for all isotopes and all types of examinations. The diagnostic reference levels were determined using these annual reports.

The 19 examinations and their DRLs:

Cerebral bloodflow	HMPAO	– 1000 MBq
	ECD	– 800 MBq
Heart function	TcCO	– 800 MBq
	Erytrocytes	– 800 MBq
Lung perfusion (MAA)	Planar images	– 125 MBq
	SPECT	– 200 MBq
Myocardial perfusion (tetrofosmin or sestamibi)		
	Stress & rest	– 1200 MBq
	Rest	– 600 MBq
	Stress	– 400 MBq
Kidney function	DMSA	– 80 MBq
Renography	MAG-3	– 200 MBq
	DTPA	– 200 MBq
Bone scan (Diphosphonates)		
	Planar images	– 600 MBq
	SPECT	– 800 MBq
Thyroid (Tc-99m)		– 150 MBq
Thyroid uptake (I-131)		– 0.6 MBq

Results from First Reports

Two years after the regulations came into force (1 October 2002) all measurements of administered activity to patients should have been done. In May 2004 a letter was sent to all hospitals with nuclear medicine departments specifying what data we were interested in, and that we wanted the data not later than 1 October 2004. Two departments were excluded; one department only examines children and the other is a dedicated PET department. A total of 32 hospitals were to report for the 19 exams.

At the end of April 2005 we are missing 9 % of the values (22/235), one hospital has still failed to report.

Values higher than the DRL was found in 20 % of the examinations (47/235).

For myocardial perfusion, mainly stress, all reporting hospitals except one exceeds DRL.

Some hospitals have motivated the high administered activity or modified their examinations (13/47).

Values less than 70 % of DRL was reported for 29 % of the examinations (68/235).

Renography, lung perfusion and bone scans are the most frequent examinations with a low reported administered activity.

The quote between maximum and minimum administered activity are for:

Bone scan 1,7
Thyroid scintigraphy 4,5
Renography with MAG3 1,7
The variation in administered activity cannot be explained by differences in equipments since the reported sensitivity for gammacameras with equivalent collimators show similar sensitivity.

Another difference between the hospitals, for the same examination, is how the individual activity is determined for each patient.

Three different methods were found:
1.All patients get the same activity (triangles)
2.The activity is adapted to the patient's weight (rhombus)
3. Random (rings)

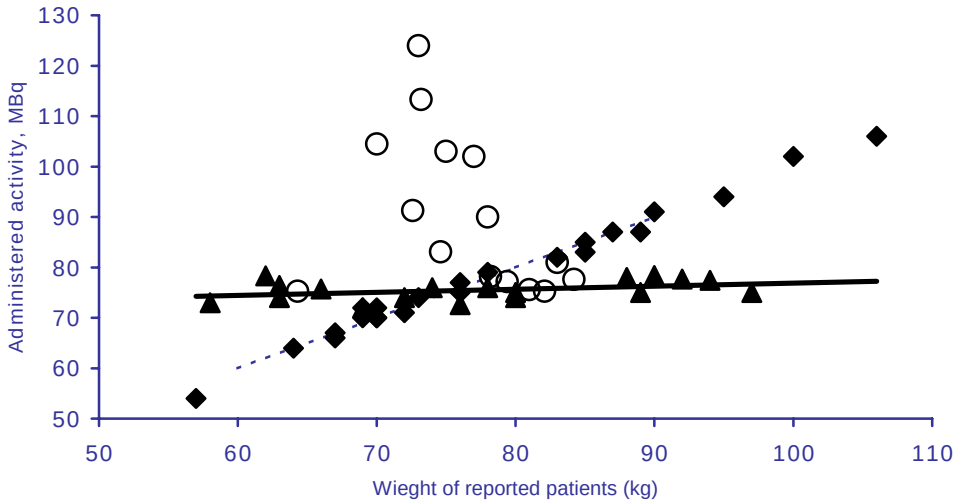

Figure 1: The graph shows reported values of measured activity for renography (MAG3) for three different hospitals (DRL = 200 MBq).

The method to adapt the activity according to the patient's weight was used in 11 % of the reported examinations, mainly for myocardial perfusion.

For 17 % of the examinations all patients were reported to have been given identical activity. This seems unlikely when activity is administered using a syringe.

The tolerance levels for the measured activity, which was administered to the patients, varied between 0 and 25 % with 10 % as the median value.

DRL in the Future

These first reports of administered activity from the hospitals and the comparison with the diagnostic reference levels show the need for changes to improve the regulations.

- A clarification of the regulations is needed since some hospitals had problems in understanding what to report.

- Heart function can be removed since not many hospitals perform this examination.
- For myocardial scintigraphy, levels should be separated for one and two day protocols.
- DRL for thyroid uptake should be for uptake measurements only and not combined with scanning.
- Activity for lung perfusion should only be reported when perfusion scan is not preceded by ventilation.
- The level for renography could be reduced.
- Examinations with increasing frequency might be added; using F-18, Cr-51 clearance and In-111 octreotide.
- The use of individual activity adapted to the patient instead of using the same activity to all patients could be included in the report.
- Other criteria for corrective measures.

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Diagnostic reference levels for paediatric radiography in Finland

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Abstract: The Radiation and Nuclear Safety Authority (STUK) in Finland has responsibility to set the national Diagnostic Reference Levels (DRLs) for the most common radiological examinations. Paediatric patients deserve a special attention due to the higher radiation risk compared with adults. The purpose was to present a method which takes into account patient size when setting DRLs in paediatrics. The overall data used in the study consisted of patient doses collected from eight hospitals: for 700 chest examinations, 100 micturating cystourethography (MCU) fluoroscopy examinations and 10 - 30 other conventional or fluoroscopy examinations. The method established by the National Radiation Protection Board (UK) for setting DRLs was found to produce extra uncertainty in the procedure and it was also troublesome to use. Reference levels for paediatric chest examinations could be given as linear curve on half logarithmic scale as a function of patient projection thickness. STUK has been collecting the data on paediatric patient doses in almost 30 Finnish hospitals as the current available data is insufficient and partly outdated. The collection is carried out by a questionnaire in which hospitals record paediatric patient doses of chest, skull, sinus, abdomen, pelvis and micturating cystourethography examinations. The DRLs will be given during the year 2005 for the most common paediatric radiographic examinations.

Introduction

Finland was 5th in the relative comparison of the amount of x-ray machines in the world at the 90's. According to the publication by Servomaa et al. (1995), more than 4.2 million x-ray examinations were done per year in Finland, from which about 9 % were done for paediatric patients. Radiation exposure due to x-ray examinations contribute about 20 % of the population's total radiation exposure, thus x-ray examinations are the biggest source of radiation after the radon radiation of the environment. Because even small amounts of radiation dose increases the risk of stochastic radiation detriments, the exposure due to radiographic examinations must be kept as low as reasonably available, as stated in the ALARA principle.

Children have many organs containing dividing cellular systems, which are especially sensitive to the ionizing radiation. Children also have a long expected life time, when mutations have enough time to create a macroscopic disease. It has been estimated that children are 2-3 times more sensitive to the radiation than the whole population. However, the paediatric patient doses are commonly much lower than those of the adult's, so the overall risk of the x-ray examinations are roughly the same magnitude for both age groups.

Those hospitals, where the average patient doses are unusually high and whose practise may have need to be improved, can be found by using the DRLs. The Medical Exposure

Directive of the European Commission (97/43/Euratom), and the Ministry of Health and Social Affairs in Finland (Decree 423/2000), regulates the Radiation and Nuclear Safety Authority (STUK) to promote the establishment and use of the DRLs for diagnostic examinations in radiology.

Patient's size is an important factor of the level of dose received from x-ray examinations. The influence of size is minimised on adults so that the mean weight of the measurement sample is close to the reference weight of 70 kg for a standard patient. Variation of size is more remarkable amongst children and the use of single reference size (as suggested by European Commission 1996) is impractical. Even dividing the children in different age groups, for example 0 - 1 years, 1 - 5 years, 5 - 10 years and 10 - 15 years, does not solve the problem, because children within the same age group can still be of very different sizes. Mainly because of the lack of the information of the doses of different sized patients, the DRLs have not yet been given for paediatrics in Finland. The aim of the study was to collect the results of earlier studies on paediatric patient doses in x-ray examinations in Finland and to present a method which takes into account patient's size when setting the DRLs.

Material and Methods

The material used in the study consisted of two research material published earlier. The first was part of an EU financed program, where paediatric patient doses were collected in four University hospitals between1994 - 1998 (Servomaa et al. 2000). The second was part of an optimization project in two children hospitals during years 1999 - 2001 (Parviainen et al. 2000). The data was supplemented by new measurements of patient doses mainly in chest examinations in two hospitals. Dose-Area-Products (*DAPs*) were measured by using calibrated DAP-meters and entrance surface doses (*ESDs*) were determined calculatorily by measuring tube yields of each x-ray machine used and by scan parameters recorded (equation 1).

$$ESD = BSF \cdot Y_{U,f} \cdot Q \cdot \left(\frac{FCD}{FSD} \right)^2 \qquad (1)$$

BSF is the back scatter factor (altered between 1,33-1,46 in chest examinations) and $Y_{U,f}$ is the measured x-ray tube output, which is depended on tube voltage (*U*) and filtration (*f*) used. *Q* is the current (*I*) time (*t*) product (practical unit mAs), *FCD* is the focus-chamber distance used in x-ray output measurements and *FSD* is the focus-skin distance in examinations.

Different methods for estimating the size of the patient have been introduced in literature. Patient's size can be estimated by dividing patients in different age groups or the weight of the patient can be used especially for neonates and small children, or the size of the patient can be estimated by different mathematical approximations, which are based on the weight and the height of the patient. Different methods were compared by comparing the standard deviations and the correlations between patient dose and the chosen size quantity. National Radiation Protection Board (NRPB) have published a method which uses normalization factors, which normalize the doses of real patients to represent the doses of five standard sized patients for whom the DRLs could be specified (Hart et al., 2000). As the Finnish x-ray examination practise differs slightly of the practise of the

Great Britain, the suitability of the method, and the approximations needed, were studied, too.

Results and conclusions

The overall data used in the study consisted of patient doses collected from eight hospitals: for 700 chest examinations, 100 micturating cystourethography (MCU) fluoroscopy examinations and 10 - 30 other conventional or fluoroscopy examinations. The data of chest examinations was sufficient for comparing different methods. The equivalent cylindrical diameter (*ECD*) calculated by a formula that includes weight and height of the patient correlated well with the patient dose. Projection thickness measured parallel with x-ray beam correlated well, also. Correlation factors for exponential curve fittings were 0.86 - 0.99. As the projection thickness is more practical to use, it will be a useful quantity when setting the DRLs.

NRPB used projection thicknesses in defining normalization factors, too. As in the original publication, grid was not used when deriving normalization factors in chest examinations, there followed error from approximating the influence of the grid. As the voltage had to be averaged to 100 kV and the total filtration differed from that used by NRPB, the total relative error was estimated to 12 % in *ESD* and 20 - 40 % in *DAP*. As the method was found troublesome to use in practise, too, the projection thickness was concluded the best method approximating the size.

Children were divided in five thickness groups in such a way that each thickness groups approximately coincided with age groups as in the NRPB:s report, too. DRLs were calculated in chest examinations as third quartile value for each thickness group. *ESD* values differed from 34 to 66 μGy in AP/PA projection and from 52 to 226 in LAT projection. *DAP* values differed from 5 to 39 $mGycm^2$ in AP/PA and from 8 to 109 $mGycm^2$ in LAT projection.

As the paediatric x-ray examinations are seldom done, it would be difficult to collect enough patient data (at least ten patients) for each thickness group. More practical method is to present the DRLs as a function of patient thickness (Figure 1). As the exponentially growing DRL-curve is fitted on half logarithmic scale, it shows as a linear line. When paediatric DRLs are presented as a curve, hospitals can compare their patient doses straight on graph and the total patient dose data needed is significantly diminished.

In order to set the DRLs there should be patient data collected from at least 20 national hospitals and the collection should include most frequent examinations done for children. In Finland, data collection has been focusing on those hospitals where paediatric patients are examined the most. That includes all five university hospitals, most of the central hospitals and some big health centres and district hospitals. The data collection has been carried out during years 2004-2005 by a questionnaire in which hospitals have recorded individual examination parameters including calculated entrance surface doses (*ESD*s) and measured dose-area-products (*DAP*s) of children under age of 16 years. As a result of the patient dose collection, DRLs will be set for paediatric patients until the end of the year 2005.

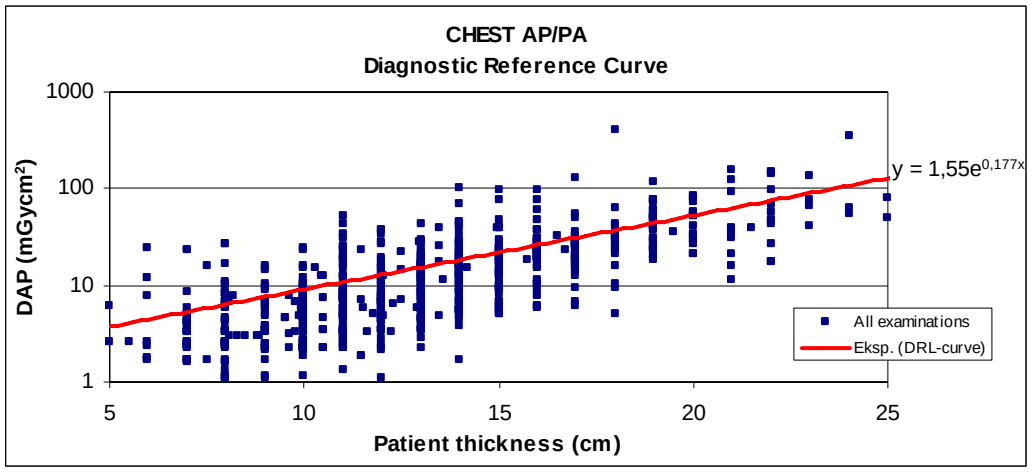

Figure 1: Diagnostic reference curve. Roughly 75 % of patient doses go under the curve.

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Trends in radiological examinations of the spine: frequencies of plain radiography, CT and MRI examinations and impact on collective effective dose in Norway

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Abstract: The present study aimed at examining possible shifts in modalities from plain radiography (CR) to Computed Tomography (CT) and Magnetic Resonance Imaging (MRI) of the spine over the last decade and assessing the potential changes in the collective effective dose from spine examinations. Nationwide surveys have been carried out to assess the numbers of radiological examinations conducted in Norway in 2002 and 1993. The results show a considerable increase of MRI and CT examination frequencies and a modest decrease of the CR frequency implying an increase of 50 % in the collective effective dose from medical imaging of the spine. Some possible explanations and consequences of the observed results are discussed.

Introduction

The development in medical imaging technology has been considerable during the last decade, with significant implications expected for both the pattern of use of the different imaging modalities and the doses delivered to patients. Computer Tomography (CT) and Magnetic Resonance Imaging (MRI) have been established as modalities for examination of the spine. A relevant question from a radiation protection point of view is whether these modalities are replacements or supplements to plain radiography (CR), and how this affects the collective effective dose.

Material and methods

The number of spine examinations applying CR, CT and MRI forms part of a larger survey which covers virtually every radiological examination conducted in Norway during 2002, chiropractic purposes not included. This larger survey is the first national survey conducted in Norway since 1993. The survey has utilised detailed information available from radiology management systems at all hospitals and clinics across Norway, regarding the number of examination codes. The code system was prepared by the Norwegian College of Radiology (2002) and nearly all hospitals and clinics had implemented this system by 2002.

The 2002 data are compared with 1993 data as obtained in a NRPA survey published by Olerud (1997). The data are comparable due to similarity in data collection. Both data sets are based on examination codes, but different versions of this code system may lead to some methodological uncertainty. In the 1993 data one examination code is assumed to represent one examination. This is not the case in the 2002 data where one examination may generate more than one code and therefore impose a minor overestimation of the examination frequencies. Codes which obviously do not represent an examination, e.g.

additional MRI series, are excluded according to the user manual prepared by the Norwegian College of Radiology (2002) which defines some codes as descriptors of the examination procedure rather than examinations. In the following, the term number of examinations refers to the number of examination codes.

The collective effective dose from spine examinations is calculated from the total number of examinations and the mean effective dose per examination type. Seven types of CR examinations and four types of CT examinations are included in the calculations. Examination of the lumbar spine is the most frequent type for both modalities and the mean effective dose ranges from 0.88 mSv per CR examination to 4.5 mSv per CT examination. The mean effective dose per examination are taken from the latest countrywide dose surveys as published by Olerud (1997) for CR and CT examinations, since frequency of a specific radiological examination type is assumed to change more rapidly than the corresponding mean effective dose.

Results

The reported total numbers and frequencies per 1000 inhabitants of radiological examinations of the spine in Norway in 1993 and 2002 are presented in table 1, which also includes the collective effective doses from CT and CR examinations. The total frequency of spine examinations has increased by 19 % since 1993. The frequency of CR examinations conducted is reduced by a factor 0.8 while the frequencies of CT examinations and MRI examinations are increased by a factor 2.6, and 14.6, respectively. In 2002, CT and MRI accounted for 23 % and 16 % of radiological examinations of the spine, respectively, while the corresponding 1993 values were 10 % and 1 %.

The increase in the examination frequency per 1000 inhabitants of CT and MRI examinations and the decrease in CR examinations of the spine from 1993 to 2002 are shown in figure 1. Figure 2 depicts the contributions to the collective effective doses (CED) from CR and CT examinations in 1993 and 2002, respectively. The total CED from spine examinations has increased by a factor 1.5 from 1993 to 2002 due to the increased CT examination frequency. In 2002, CT accounted for 59 % of the annual collective effective dose from examinations of the spine as opposed to 32 % in 1993.

Table 1. The reported number of radiological examinations of the spine (in 1000 examinations) per imaging technology, the examination frequency per 1000 inhabitants and the corresponding contributions to the collective effective doses from CR and CT examinations in Norway in 1993 and 2002.

	No of examinations ($\times 10^3$)				**Frequency (per 1000 inhabitants)**				**CED per modality (man Sv)**		
Year	**CR**	**CT**	**MRI**	**Total**	**CR**	**CT**	**MRI**	**Total**	**CR**	**CT**	**Total**
2002	294	109	76	479	65	24	17	105	325	468	793
1993	333	40	5	378	77	9	1	88	365	175	540

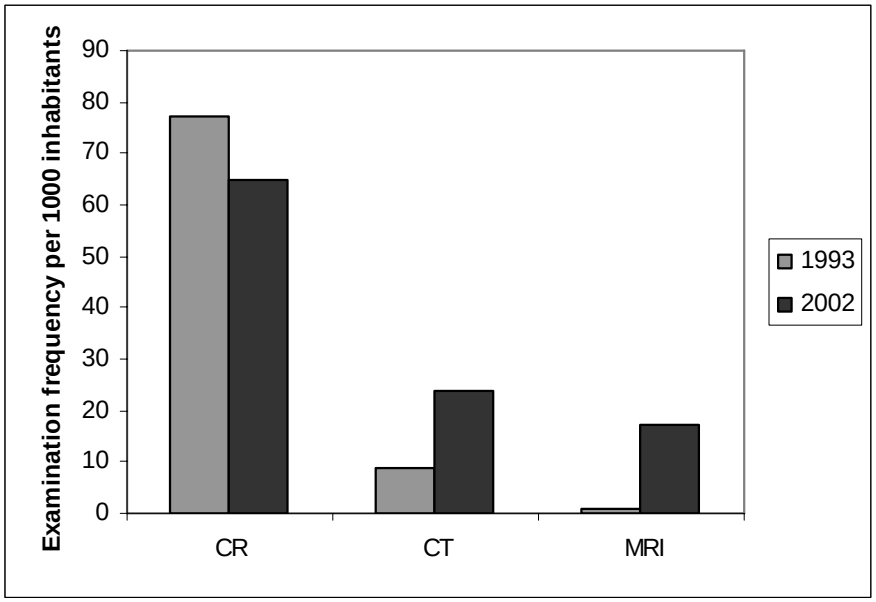

Figure 1. Frequencies of CR, CT and MRI examinations of the spine in 1993 and 2002 per 1000 inhabitants.

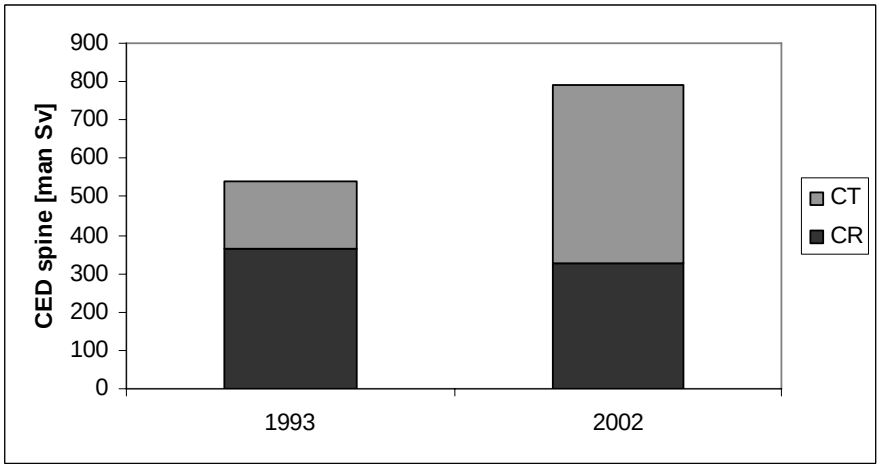

Figure 2. The contribution to the collective effective dose (CED) from CT and CR examinations of the spine to the Norwegian population in 1993 and 2002.

Discussion and conclusion

Within a decade (1993-2002) the frequency of CR examinations of the spine is slightly decreased, while the frequency of CT and especially MRI examinations is substantially increased. It is unlikely that the observed overall increase in frequency of spine examinations of 19 % reflects increase in morbidity in the population.

In order to assess whether the findings are affected by methodological uncertainties as a result of applying the number of examination codes in order to calculate the frequencies, the ratio of the number of procedure codes, e.g. administration of intra vascular contrast media, to the total number of codes was explored. This yielded maximum possible over estimations of 35 %, 12 % and 9 % in the CT, CR and MRI frequencies respectively, i.e. the frequency changes in CT, MRI and CR are at least factors 1.7, 13.4 and -0.7, respec-

tively. The estimated increase in the MRI frequency is carried out in a conservative way, since all the additional MRI series are excluded from the material.

A question that arises is why the increase in CT and MRI frequencies not has lead to a corresponding decrease in the CR frequency. The need for differential diagnosis might offer an explanation, and in this case the consequences for patient doses and public expenditures are not difficult to accept. Uncertainty of examination indications is another explanation. An audit of referrals from Norwegian general practitioners for plain radiography of the lumbosacral spine carried out by Espeland and co-workers (1999) showed "that a great proportion of the referrals do not conform to recent clinical recommendations", and furthermore that these referrals "yield few clinically relevant findings and could probably be cancelled".

The observed trends in which the frequency of CR examinations is lowered due to the introduction of the competing modalities CT and MRI confirm an increased perception of that the latter modalities can offer safer diagnosis and better patient outcome. Justification of consequences of this trend with respect to public expenditures and increased collective effective dose due to CT examinations relies on the correctness of this perception. Unfortunately also for MRI the association between found anatomic irregularities in the lumbar spine, clinical diagnosis and patient outcome is controversial (Lurie 2003).

From a radiation protection perspective the pronounced trend towards non-ionising examinations (MRI) is promising. A retrospective study by Nyquist and co-workers (2004) covering 20 years in a Norwegian referral hospital showed significant dose savings due to an intended policy to select MRI instead of CT indicates a possibility for further reduction of the collective effective dose from spine examination in the future.

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Diagnostic reference levels for nuclear medicine examinations in Finland

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Abstract: Implementation of diagnostic reference levels (DRLs) into national legislation was done by the Decree on the Medical Use of Radiation (423/2000) issued by the Ministry of Social Affairs and Health. The DRLs were introduced for nuclear medicine examinations in Finland in 2000 by STUK - Radiation and Nuclear safety Authority. In this paper the national reference levels for nuclear medicine examinations are compared to the actual administered activities based on the results of the surveys on the use of radiopharmaceuticals in 2000 and 2003 in Finland. The possible influence of the DRLs on actual administered activities and the need of revision of the DRLs are discussed.

Introduction

STUK issued the first national DRLs for 23 nuclear medicine examinations in its decision no. 596/310/00 (8 December 2000), and these will be revised as necessary. Administered activities in paediatric examinations are fractions of activities given to adults and determined as a function of the body weight. In addition, minimum activities to be administered to children for some radiopharmaceuticals were also given. The decision also gives information on the practical use of DRLs. The regulatory guide ST 6.3 Use of radiation in nuclear medicine given by STUK in 2003 specifies information on the use of the DRLs.

The first national DRLs were given on the basis of the results of the national survey made by STUK concerning the use of radiopharmaceuticals in 1997 in Finland as well as of the international recommendations. The reference levels were first set for examinations the amount of which was large and which therefore caused a high collective dose, and for examinations which cause a high dose to an individual patient. The DRLs are related to average activities for groups of standard size patients and to typical examinations with standard equipments. The DRLs set are not optimal activities for a specific procedure and equipment at local level.

Results and discussion

After setting the DRLs the country wide survey on the use of radiopharmaceuticals has been made twice (2000 and 2003).

Table 1. are shown the diagnostic reference levels given in 2000, the mean administered activities (weighted with the number of examinations) and the range of the mean activities used in different hospitals in years 1997, 2000 and 2003.

At first the aim of the use of the DRLs was to eliminate the highest and lowest activities used. When considering the ranges of the mean administered activities in different examinations in different hospitals in years 1997, 2000 and 2003, no remarkable influence

on the mean administered activities is to be seen. There still are differences between different hospitals in the administered activities.

Some needs to reconsider the DRLs given in 2000 came out:

- The number of examinations using ^{18}F-FDG made both with PET cameras and gamma cameras with coincidence system is increasing. The DRL has to be set for those examinations.
- In the national survey from 1997 the examinations to adults and to children were not separated. When giving the DRLs for some examinations (e.g. glomerulus filtration rate studies) it was not taken into account that the majority of the examinations are made to children and only a minority to adults. This led to a too small DRL for adults.
- Some new radiopharmaceuticals have been introduced. It may be necessary to set the DRLs for some new radiopharmaceuticals. At the same time the use of some other radiopharmaceuticals is decreasing. The annual number of 5 examinations to which the DRLs were given in 2000 was in 2003 less than 100. The DRLs for these examinations may not be necessary.
- In 2003 10 most frequent examinations constituted about 80 % of all nuclear medicine examinations performed in Finland. It may not be necessary to set the DRLs to other than those 10 most frequent examinations.

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Table 1. Diagnostic reference levels for adults, the mean activities and the range, and the numbers of examinations in 1997, 2000 and 2003.

Examination	Radiopharmaceutical	DRL (MBq)	Mean ad. activity and range in 1997 (MBq)	Mean ad. activity and range in 2000 (MBq)	Mean ad. activity and range in 2003 (MBq)	Number of examinations in 1997/2000/2003
Bone and soft tissue						
Bone imaging, planar	^{99m}Tc-phosphates and phosphonates	600	595 (100-740)	627 (370-740)	633 (500-740)	18767/17835/15993
Bone imaging, SPECT	^{99m}Tc-phosphates and phosphonates	800	727 (500-749)	716 (550-740)	675 (500-800)	1646/396/436
Infection imaging	^{99m}Tc-leucocytes (HMPAO)	300	234 (180-600)	238 (120-600)	245 (150-400)	969/878/911
	^{111}In-leucocytes	20	19	18	18 (18-18.5)	238/365/361
Lungs						
Lung perfusion	^{99m}Tc-MAA	150	110 (50-222)	104 (50-185)	125 (50-170)	5986/5056/4376
Lung ventilation	^{99m}Tc-aerosol (Technegas)	50	45 (26-75)	37 (20-100)	41 (20-100)	425/1217/2747
	^{99m}Tc-aerosol (Venticoll)	80	62 (11-100)	69 (11-111)	68 (11-111)	810/474/96
Gastrointestinal tract						
Meckel's diverticulum	^{99m}Tc-pertechnetate	400	233 (40-560)	195 (40-500)	311 (185-555)	92/64/34
Bile duct scan	^{99m}Tc-IDA compounds	150	124 (37-370)	155 (74-240)	133 (74-190)	101/110/73
GI blood loss	^{99m}Tc-erythrocytes	600	603 (450-740)	610 (400-740)	573 (400-800)	71/51/33
Bile salt absorption	^{75}Se-HCAT	0.4	0.37	0.37	0.37	230/277/179
Kidney, urinary system						
Renal imaging	^{99m}Tc-MAG3	150	148(80-370)	106 (70-370)	117 (70-185)	2900/2822/2059
	^{99m}Tc-DTPA	300	166 (74-370)	223 (37-370)	183 (111-370)	2409/1260/683
Cardiovascular						
Myocardial perfusion, SPECT	^{201}Tl-ion	100	99 (74-111)	97 (74-111)	104 (74-105)	2179/1646/969
	^{99m}Tc-MIBI	1 000[1]	1106 (1 000-1 150)[2]	1 015 (980-1 190)[2]	939 (523-1 140)[2]	1602/956/743
	^{99m}Tc-tetrofosmin	1 000[1]	905 (820-925)[2]	984 (740-1240)[2]	973 (740-1 200)[2]	649/2191/2908

Cardiac function (equilibrium)	^{99m}Tc-erythrocytes	700	604 (500-800)	626 (400-740)	595 (500-740)	191/307/334
Brain						
Cerebral perfusion, SPECT	^{99m}Tc-HMPAO	740	733 (600-900)	716 (500-740)	802 (500-900)	751/504/316
	^{99m}Tc-ECD	740	740	569 (550-740)	576 (550-740)	330/50/65
Thyroid, parathyroid						
Thyroid imaging	^{99m}Tc-pertechnetate	150	122 (74-185)	123 (74-185)	125 (71-185)	370/218/152
	^{123}I-iodide	20	12 (7.4-40)	11 (7.4-25)	13 (7.4-25)	134/186/92
Thyroid metastases (after ablation)	^{131}I-iodide	400	171 (10-370)	230 (74-370)	271 (75-400)	561/530/633
Parathyroid imaging	^{99m}Tc-MIBI	740	627 (300-740)	740	798 (700-900)	102/14/286

[1] For combined rest-exercise protocols carried out on the same day the total activity should not exceed 1000 MBq for SPECT.
[2] Some of the examinations were not carried out on the same day.

A method for establishing diagnostic reference levels for paediatric patients

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Abstract: Two difficulties are encountered in establishing diagnostic reference levels (DRLs) for paediatric patients: the patient doses to be measured are quite small, and the range of patient sizes is very large. We have determined the patient doses from exposure settings recorded at the actual examination combined with radiation outputs measured for each tube/generator combination used. The recorded doses are compared across the range of patient sizes using equivalent diameters calculated from height and weight of the patient. Based on the recorded doses, DRLs will be established as functions of equivalent diameter.

Introduction

Diagnostic reference levels in radiology have been introduced over the past 15 year. The diagnostic reference level (DRL) for an examination is defined in the Council Directive 97/43/Euratom (1997) as the dose level for typical examinations for groups of standard-sized patients with different types of equipment. DRLs have been introduced in the Nordic countries first through recommendations from The Nordic Radiation Protection Co-operation (1996). Later DRLs have been incorporated into the legislation, e.g. the regulations of the Danish National Board of Health (1998). In the hospitals, patient doses must be measured for a range of standard-sized patient, and the average dose is not to exceed the DRL.

However, DRLs have been set up mainly for adult patients. Suggestions for DRLs for paediatric patients have been published in European guidelines on paediatric radiology (Kohn et al. (1996)), and by the National Radiological Protection Board in the UK (Hart et al. (2000)). Due to the wide range of sizes in paediatric patients, several reference levels have to be given for each examination. Paediatric DRLs can be classified either according to the age of the child or to the size of the child, i.e. height and weight.

In this report we describe the current project for establishing paediatric DRLs in Denmark. Establishment of DRLs is based on determination of actual patient doses for a large number of patients in a range of hospitals. When the distribution of doses has been determined, the DRL is usually set at the third quartile, i.e. 75% of the measured doses should be below the DRL.

Determination of individual patient doses

In cooperation with 5 Danish hospitals we are measuring patient doses for a number of different examinations of paediatric patients. In a pilot project, doses were measured as dose area products (DAP). However, it turned out that there was a charge-build-up in the DAP meters between exposures, giving rise to errors in the dose readings. These errors

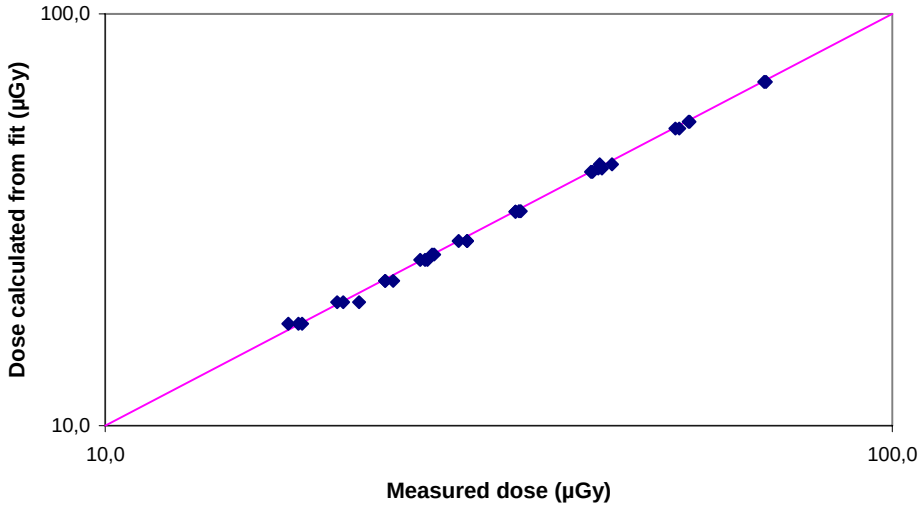
are insignificant for measurements on adult patients where both the doses and the field sizes are larger, but very significant for measurements on paediatric patients. Use of thermoluminescence dosimetry (TLD) tablets was then considered, but they would not be sensitive enough to the low doses used, especially for thorax examinations. More sensitive TLD tablets are available but could not be used here since the measurements where done at 5 different locations over an extended time period. The more sensitive TLD tablets must be used within a short time period after annealing, and this was not feasible in the current study.

It was finally decided to determine the patient doses as entrance surface doses (*ESDs*) calculated from exposure parameters recorded at the examination and output parameters determined for the equipment used, similar to methods described by Faulkner et al. (1999) and Harju et al. (2003). For each set of equipment, doses free in air at a known distance from the x-ray tube focal point (focus-to-detector distance, *FDD*) were measured for different combinations of high-voltage (kV) and current-time product (mAs), and the data was fitted in a two-step process. First, the measured doses for a given high-voltage setting were fitted as a function of current-time products assuming proportionality. These output factors (µGy/mAs) were then fitted as second order polynomials in high-voltage. One set of fitting parameters (*a*, *b* and *c*) is thus obtained for each equipment. Doses for the individual examinations can then be determined from:

$$ESD = (a \cdot V^2 + b \cdot V + c) \cdot Q \cdot \left(\frac{FDD}{FSD} \right)^2 \cdot BSF \,, \qquad (1)$$

where *Q* is the current-time product, *FSD* is the focus-to-skin distance and *BSF* is the backscatter factor.

Figure 1 shows an example from one set of equipment of a plot of doses calculated from the fitting parameters (without backscatter factor) versus the measured doses, the maximum derivation is 5%. Harju et al. (2003) have previously reported an uncertainty of about 30% when using common fitting parameters across a range of different equipment. Using one measured dose value for known radiation quality reduces the uncertainty be-



Figur 1. Double-logarithmic plot of calculated doses versus measured doses for one set of equipment. The line served only as a guide to the eye and indicates full correspondence.

low 10%. The method used here where separate parameters are determined for each set of equipment thus gives a lower uncertainty. The uncertainty in the determination of actual entrance surface doses for the paediatric patients will be slightly higher due to the uncertainty in determination of the FSD, but the total uncertainty using our method will still be about 10%.

For each patient, sex, age, height, and weight are recorded. It is desirable to describe the variation of patient size using a single parameter, and for this we have chosen the equivalent diameter as defined by Lindskoug (1992):

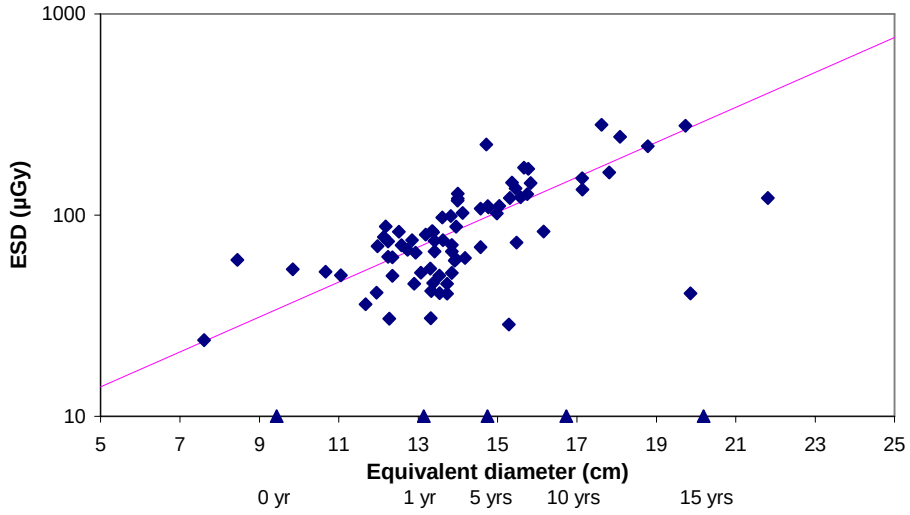
$$d_{eq} = 2 \cdot \sqrt{\frac{m}{\pi \cdot h \cdot 1\,g\,cm^{-3}}}\,, \tag{2}$$

where m is the weight, and h is the height of the patient.

For reference, we have also calculated equivalent diameters for standard paediatric patients: newborns, 1-year, 5-years, 10-years and 15-years old. Height and weight for these standard patients were taken from growth curves from The Danish Paediatric Society (2004). The equivalent diameter for a standard patient is taken as the average of the values obtained for boys and girls, respectively, using the medians of height and weight at a given age.

Determination of diagnostic reference levels

The calculated ESDs for a given examination at a given hospital are plotted as a function of equivalent diameter. Hart et al. (2000) showed that since the exit dose ("film" dose) should be kept constant, the ESD should increase nearly exponentially with the thickness of the patient, approximated here by the equivalent diameter. Thus, the *ESDs* are fitted as exponential functions of the equivalent diameter. An example of preliminary data is given in figure 2. The equivalent diameters for standard patients are given in the figure for reference.



Figur 2. Example of plot of ESD as a function of equivalent diameter (rhombs). Equivalent diameters of standard patients are indicated on the bottom axis (triangles, see text).

Similar plots will be constructed for all participating hospitals, and the fits can be interpreted as average doses for standard patients as a function of the equivalent diameter. From these functions, DRLs for paediatric patients for the given examination will be obtained, again as a function of equivalent diameter. In addition, DRLs for the standard-sized patient will be provided as a quicker reference.

Use of the paediatric DRLs

When patient doses for paediatric patients are measured in a hospital, the measured doses can be plotted as function of the equivalent diameter of the patient, in a way similar to figure 2. In this case, the curve for DRL as a function of equivalent diameter should be included in the plot. If the majority of the measured doses are below the DRL curve, and no single doses are far above the DRL curve, the patient doses are in compliance with the diagnostic reference levels.

Acknowledgement:

We thank the Nordic working group for medical x-ray diagnostics for helpful discussions on the method and preliminary results.

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Experience from the first compilation of diagnostic standard doses from x-ray examinations in Sweden

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Abstract: The concept of diagnostic reference levels (DRL) has been introduced into Swedish legislation. Numerical values have been established for twelve x-ray examinations and all Swedish hospitals are required to assess the diagnostic standard doses for all x-ray stands where these examinations are conducted. The diagnostic standard doses shall be compared with the DRL. The hospitals were required to send data to the Swedish Radiation Protection Authority (SSI) before 1 October 2004. In this paper a first evaluation of the collected data is presented.

SSI has compiled more than 1300 diagnostic standard doses from the hospitals. Between 0 and 19 % of the various examinations are exceeding diagnostic reference levels and for these dose reduction measures have to be performed. The analysis shows that the regulation has to be revised. The numerical values have to be altered and the method of assessing the diagnostic standard dose has to be improved. The information to the users must be improved.

The concept of diagnostic reference levels

As a consequence of the medical exposure directive (1) the concept of diagnostic reference levels have to be establish in all member states of the EU. However, the methodology might differ substantially between different countries, e.g. the Nordic countries have chosen different approaches.

The purpose of introducing diagnostic reference levels is to identify practices where patients undergoing x-ray examinations receive unnecessary high radiation doses, and to encourage the users to investigate the cause and if possible reduce the radiation dose. In addition, further effects could be expected, such as

- increase of the awareness of radiation dose to the patients in the hospital
- encouragement of collaboration and comparisons between hospitals
- support of actions for dose reduction performed by the regulatory bodies

The diagnostic reference levels should facilitate the optimization process. Optimization is the most important task in radiation protection in diagnostic radiology.

Diagnostic reference levels in diagnostic radiology in Sweden

Diagnostic reference levels were introduced in Sweden in 2002 (2) for a number of specified x-ray examinations in diagnostic radiology (Table 1-3). When selecting the quantities for the diagnostic reference level due consideration was given to practical aspects for the users. It should be easy for the hospitals to assess the diagnostic standard dose and to compare it with the DRL. All hospitals performing at least 100 examinations per year for a specific investigation in a specific x-ray room have to determine the diagnostic standard dose for that examination

According to the regulations the hospitals had to determine the diagnostic standard doses before 1 October 2004, two years after the regulation was issued. In February 2003 SSI specified which data had to be reported to SSI with the deadline 1 October 2004. This deadline was extended to 31 January 2005 because most hospitals stated difficulties with the collection of the diagnostic standard doses on time. SSI has now started to analyze the data.

Results from the first compilation

Data for more than 1 300 diagnostic standard doses together with essential technical/physical parameters and information about the patient were sent to SSI. The data were stored in a database (Microsoft access) and comprise some 27 000 patient examinations. Some general conclusions can be drawn from the first evaluation of the data.

A majority of the hospitals have reported great difficulties in organizing and carrying out the measurements within the time limit. Some of the hospitals have not realized that all the measurements had to be completed before the deadline for submission and they were reminded.

Problems assessing diagnostic standard dose were also observed. For conventional examinations the importance of the patient size was not always taken into account. The mean weight of the patient group was not considered to a proper extent. In computed tomography it was not always recognized that the sum of the dose-length products for all scans used for the patient examination should be calculated, and not the mean value for single scans.

The number of diagnostic standards doses exceeding the diagnostic reference level varied between 0 % and 19 % for the different examinations (Table 1-3). For coronary angiography not a single diagnostic standard dose is exceeding the diagnostic reference level. This is an example for how important it is to have sufficient data from different hospitals when establishing diagnostic reference levels. The value for the diagnostic reference level for coronary angiography was based on doses from one hospital only.

Need for improvements in the future

There is large evidence that the implementation of diagnostic reference levels will have great impact on the radiation dose to the patients. However, as anticipated the present concept and the regulations can be improved. This could be summarized as follows:

- Most values for diagnostic reference levels should be lowered
- SSI must improve the guidance for the users on how to assess the diagnostic standard dose. The comparison of diagnostic standard doses with the diagnostic reference level and other standard doses will then be more reliable.
- The users must be informed about the importance of recording certain parameters in connection with the measurements.

With the suggested improvements the concept of diagnostic reference levels will be an even more powerful tool for the optimization process.

Table 1. Diagnostic reference levels, total number of diagnostic standard doses in the database and number of standard doses exceeding the DRL (%) for conventional x-ray examinations

Conventional x-ray investigations	Number x-ray rooms	Dose area product	
		DRL (Gy*cm2)	> DRL
Heart and chest, chest health check-up	187	0,6	16 %
Coronary angiography (one or several vessels)	36	80	0 %
Barium enema with double contrast	78	50	5 %
Urography with urethra compression	85	20	16 %
Lumbar spine and SI-joints	198	10	11 %
Pelvis, hip joints (only PA/AP view)	209	4	3 %

Table 2. Diagnostic reference levels, total number of diagnostic standard doses in the database and number of standard doses exceeding the DRL (%) for computed tomography

Computed tomography	Number	CTDIvol		Dose-length product	
		DRL (mGy)	> DRL	DRL (mGy*cm)	> DRL
Brain	130	75	18 %	1200	14 %
Abdomen	113	25	1 %	N/A	NA
Thorax/Lungs	114	20	1 %	600	9 %
Lumbar spine	32	55	12 %	600	19 %

Table 3. Diagnostic standard doses exceeding the DRL in mammography (%)

Mammography	Patient measurement		Phantom measurement	
	AGD per exposure	AGD per examination	AGD per exposure	AGD OD =1
Mammography, complete examination	6 %	16 %	1 %	16 %
Mammography, screening	13 %	13 %	0 %	17 %

The total number of diagnostic standards doses is 90 screening and 67 complete examinations.

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Local skin absorbed doses in endovascular neurosurgery measured with KAP-meters and TL-dosimetry

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Abstract. The aim of this work was to assess patient-radiation risk related dosimetric quantities such as the kerma-area product, KAP, and absorbed doses to the patient's skin in endovascular neurosurgery (embolisations and coiling) in order to predict the frequency of potential acute radiation injuries. A Siemens dual image-intensifier x-ray fluoroscopy unit was used. KAP and measured skin doses were recorded using transmission ionisation chambers and thermo-luminescence dosimeters, TLD, ($Li_2B_4O_7$). The calculated local absorbed dose values, predicted by the x-ray unit at the patient's reference plane, was compared to the measured absorbed doses and with the KAP. The median and 95%-percentile KAP (correlated to the risk of late radiation effects) was 98 and 237 $Gycm^2$ (coiling) and 140 and 414 $Gycm^2$ (embolisation), respectively. The corresponding calculated (predicted) skin dose (correlated to acute radiation effects) was 1.2 and 2.8 Gy (coiling) and 1.3 and 4.0 Gy (embolisation), respectively. However, the measured median and 95%-percentil absorbed skin dose in the same interventional procedures were significantly lower 0.84 Gy and 1.6 Gy, respectively, indicating that the predicted skin dose from the x-ray unit overestimates the maximum skin dose. The conversion factor between skin dose and KAP-value, ESD/KAP, based on the predicted skin dose was 10.3±3.0 $mGy/Gycm^2$ but significantly lower 4.4±1.2 $mGy/Gycm^2$ based on the measured skin dose. Hence, in order to obtain a local skin dose of 3 Gy (corresponding to temporary skin epilation), a KAP-value of approximately 750 $Gycm^2$ is required. No immediate risk of temporary epilation for the majority of patients thus exists.

Introduction

Acute radiation injuries have been reported after radiological interventional procedures (ICRP 85 2000). It is therefore necessary to regularly measure patient doses and to have proper protocols to minimize local skin doses and skin erythema as a results of these, sometimes lengthy procedures. Several papers present data on measured absorbed dose in the patient skin (Theodorakou and Horrocks 2003, Bergeron et al 1994) and energy imparted to the patient (Gkanatsios et al 2002) following interventional neuroradiology in order to correlate to acute and late radiation effects, respectively. In this paper we report on measured local skin doses on the patients head using TLD and compare these values to the predicted skin dose given by the x-ray unit and with the measured total KAP-value.

Materials and Method

Interventional procedures Cerebral digital subtraction angiography (DSA) was performed in the vascular territories of interest for diagnosis, and used for coiling of aneurysms and embolisation of cerebral arteriovenous malformations or tumours. The routine film rate for each plane is 2 frames/second for 3 seconds (or 3 frames/second in the pres-

ence of a lesion that includes arterio-venous fistulas), then 1 frame/second for up to 10 seconds. Rotational angiography is performed when necessary (usually for 3D depiction of aneurysms) with 200 frames over 210 degrees.

X-ray equipment and patient dosimetry A Siemens Axiom Artis BA Biplanar x-ray unit, with one floor-mounted and one ceiling-mounted x-ray tube each with a 33 cm diameter image intensifier, was used. Each unit has a kerma-area product (KAP) meter mounted on the x-ray tube collimator (PTW Diamentor). The unit presents a calculated (predicted) dose at a given distance from each x-ray tube as an indicator of local skin dose. This value is calculated based on the KAP-value and known distance between the x-ray tube and the patient's reference plane.

Since each intervention is different and comprise of many projections, including both fluoroscopy and digital fluorography, it is likely that this predicted value does not correspond with the measured absorbed dose on the skin. Therefore eight pairs of thermoluminescence dosimeters, TLD, ($Li_2B_4O_7$) were positioned on the patient's head using a headband on 32 examinations including interventions. The TLD were calibrated to measure the superficial personal dose equivalent, $H_p(0.07)$ using ISO/IEC N-40 spectrum. The calibration was traceable to the national dosimetry standard lab at SSI, Stockholm. For simplicity the quantity $H_p(0.07)$ was here denoted entrance skin dose ESD_{TLD}. Three TLD were positioned on the left side of the head; two on the back of the head; one on the right side and one on each eye to estimate the dose in the eye lens. The average value of the two TL-dosimeters of the pair was used to represent the skin/eye dose at each position.

The measured maximum skin dose value, $ESD_{TLD,max}$ was compared with the unit's prediction of the maximum skin dose, ESD_{pre} by calculating the ratio $ESD_{pre}/ESD_{TLD,max}$ for each patient.

The energy imparted to the patient, ε, was estimated from the measured KAP-value and Monte Carlo calculated conversion factor, C_ε, (Alm Carlsson et al 1984) between energy imparted and total KAP-value. C_ε = 150 kg/m^2 was used. The energy imparted was computed using $\varepsilon = KAP \cdot C_\varepsilon$. Since the KAP-value is readily available after each procedure, it is useful to derive conversion factors between measured maximum skin dose and the total KAP-value in order to estimate the local maximum skin dose. Therefore conversion factors $C_m = ESD_{TLD}/KAP$ and $C_c = ESD_{pre}/KAP$ were derived.

Results

Patient dose survey Table 1 summarises the patient doses for approximately 200 procedures. Given the data in table 1 and figure 1 and 2 suggests that the risk of temporary skin epilation (ESD>3Gy) is not as high as initially assumed based on the calculated skin doses predicted by the x-ray unit.

Conversion factors Figure 1 shows the conversion factor between the measured ESD_{TLD} and KAP-value (mean±1SD=4.4±1.2 mGy/Gycm^2) and between the calculated (predicted) ESD_{pre} and KAP-value (mean±1SD=10.3±3.0 mGy/Gycm^2). The conversion factor based on the measured skin dose is thus much lower. Figure 2 show the ratio between the maximum skin dose measured by TL-dosimeters for each procedure and the maximum skin dose as predicted by the x-ray unit. The ratio is in all cases less than one (mean±1SD = 0.47±0.18) but varies considerably indicating a poor correlation between the two.

Table 1. The table show the median (50%-percentile) and 95%-percentile of the fluoroscopy time, KAP-value, predicted (calculated) entrance skin dose from both the floor- and ceiling-mounted x-ray unit, the measured entrance skin dose and the energy imparted, ε.

Examination, percentile	Exams	Fluoroscopy time, (min)	KAP_{flo} ($Gycm^2$)	KAP_{cei} ($Gycm^2$)	KAP_{total} ($Gycm^2$)	$ESD_{pre,flo}$ (Gy)	$ESD_{pre,cei}$ (Gy)	ESD_{TLD} # (Gy)	ε (J)
Angiography	110								
50%		6.2	30	10	40	0.34	0.13	0.13	0.6
95%		35.7	97	25	123	1.27	0.39	0.45	2.0
Coiling	61								
50%		30.6	72	25	98	1.2	0.70	0.68	1.5
95%		79.2	180	57	237	2.8	1.9	1.5	3.7
Embolisation	25								
50%		35.2	103	37	140	1.3	0.63	0.84	2.2
95%		63.2	295	119	414	4.0	1.9	1.5	5.9

The ESD_{TLD} is based on 32 procedures.

Measured skin dose distribution Figure 3 shows the 95%-percentile skin dose in each position for the three procedures. In the coiling procedure, the highest skin doses are located on the patient's left side and on the back of the head. The median and 95%-percentile measured maximum skin dose per interventional (coiling or embolisation) procedure was 0.8 and 1.5 Gy, respectively. The absorbed doses to the patient's left eye (which was typically higher than the right eye) were comparably lower with a median of 0.030 Gy and 95%-percentile of 0.38 Gy. The maximum dose in the eye was 0.51 Gy.

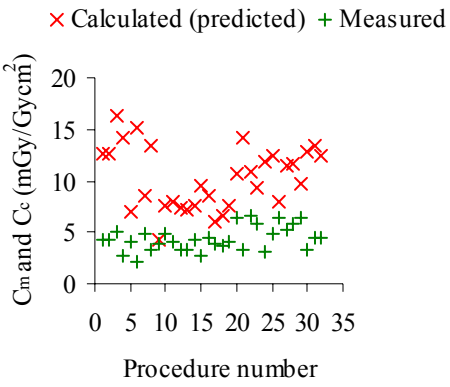

Figure 1. The figure shows the conversion factors C_m (measured) and C_c (calculated) between ESD_{TLD}, ESD_{pre} and KAP, respectively.

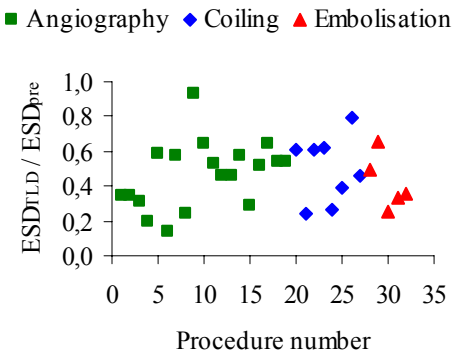

Figure 2. This figure shows the ratio between the measured ESD_{TLD} and calculated (predicted) ESD_{pre} for the three procedures.

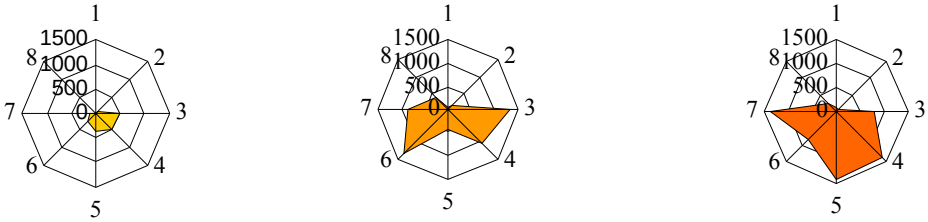

Figure 3. The figures show the angular distributions of the 95%-percentile of measured ESD_{TLD} in angiography (left), coiling (middle) and embolisation (right). Positions 1 and 8 are the right and left eye; position 2 are on the right side; positions 3 and 4 on the right and left side on the back; positions 5-7 on the left side of the head.

Discussion

Theodorakou and Horrocks (2003) measured skin doses using TLD as well as KAP for thirty cerebral embolisations. Their ESD_{TLD} (median 0.6 Gy, max 3.4 Gy) and KAP (median 40 $Gycm^2$, max 321 $Gycm^2$) are similar to ours and also indicate a good linear correlation between ESD_{TLD} and KAP (r^2=0.79-0.86; ours r^2=0.93). Bergeron et al (1994) also reports absorbed doses in accordance with ours (median KAP=96 $Gycm^2$ and median ESD_{TLD}=0.47 Gy). Gkanatsios et al (2002) estimated the energy imparted to the patient and their median value (6.7 J) was higher than reported here (see table 1).

Conclusions

The conversion factor between measured skin dose and KAP was 4.4±1.2 $mGy/Gycm^2$, which is lower than the corresponding factor given by the x-ray unit. No immediate risk of temporary epilation (ESD>3Gy) for the majority of patients thus exists.

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Estimations of effective dose in X-ray examinations derived from information stored in PACS

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Abstract: Information about each X-ray examination, in a modern digitized X-ray department is generated and stored in a PACS. Appropriate conversion factors, e.g. E/DAP, can be applied to separate projections and summed to the total effective dose for each examination. The objectives of the work were (i) to investigate the accuracy and precision in the calculated effective dose (ii) to identify data for registration of (1) patient dose, (2) exposure data, and (3) patient information (iii) to make it possible to derive dose statistics on patient level for documentation of diagnostic standard doses, optimizations, constancy checks, and future epidemiological studies. The effective doses were calculated using Monte Carlo based computer programs or by using tabulations. Conversion factors were calculated for different levels of information and the individual effective dose was compared to the most precise estimation. The results suggest that the accuracy in the estimations of effective dose increases by added information about the patient (gender, size) and how the examination was performed.

Introduction

Information about each X-ray examination, in a modern digitized X-ray department, is generated and stored in a picture archive and communication system, PACS. Quantities as DAP (dose area product), $CTDI_{vol}$ (computed tomography dose index), DLP (dose-length product) or AGD (average glandular breast dose) can be registered which makes it possible to estimate effective dose individually. The diagnostic standard dose can then be calculated as the average of all patients rather than, as today, from a sample of 20-30 patients. Appropriate conversion factors, e.g. effective dose to dose area product, E/DAP, can be applied to separate projections and added up to the total effective dose for each examination. This is possible when specific information about each examination is stored in the image file.

The objectives of the work were (i) to investigate the accuracy and precision in the calculated effective dose for each examination using the stored information, (ii) to identify appropriate and necessary data for registration of (1) patient dose (e.g. DAP, AGD, $CTDI_{vol}$, DLP), (2) exposure data (e.g. voltage, filter, distances) and (3) patient information (e.g. age, gender, thickness), (iii) to make it possible to derive dose statistics on patient level for documentation of diagnostic standard doses, optimizations, constancy checks, and future epidemiological studies.

Material and methods

For conventional radiography, the conversion factors and effective doses were calculated using a Monte Carlo based computer program (Tapiovaara et al, 1997). For CT a MS Excel spreadsheet system (ImPACT) and dose data simulated by Jones and Shrimpton (1993) were used. The effective dose was calculated according to ICRP 60 (1991). The AGD values for mammography were calculated from entrance surface air kerma, ESAK, according to Dance et al (1990, 2000). Examination information from radiography (spine, lung, urography, pelvis), mammography, and CT examinations were gathered from the county councils of Östergötland and of Jönköping. Examination types were chosen in reference to those having stipulated diagnostic reference level by the Swedish Radiation Protection Authority, SSI (FS 2002:2). Meta data from the DICOM (Digital Imaging and Communications in Medicine) header (NEMA, 2003) were extracted. Conversion factors were calculated for different levels of information and the total individual effective dose was compared to the most precise estimation.

As starting positions in the comparisons of accuracy, conversion factors suggested by SSI in the comments to FS 2002:2 (2002) were used. For conventional radiography, these conversion factors (E/DAP) originate from Nordic guidance levels (1996), an adaptation from Hart et al (1994). The European Commission guidelines provide the corresponding CT (E/DLP) conversion factors in EUR 16262 (1999) and for mammography (AGD/ESAK) in EUR 16262 (1996).

Conversion factors for the radiographic examination types were calculated using increasing amount of information categorized into four levels. Level 1 uses the conversion factors provided by SSI (FS 2002:2, comments). The effective dose was calculated using the total DAP value for the examination. In level 2, information about each specific projection was utilized and the effective dose for the complete examination was calculated using the separate DAP values. Information of beam quality (tube potential, filtration) was added in level 3. Level 4 utilizes the patient sex and thickness. The thickness (radiation length through the patient) can be derived from the tube loading in combination with specific measurements of the automatic exposure control feature, AEC. Dose values from each level were compared to the best estimation possible, using all applicable data from each projection (24 to 100 different patients per type of examination).

The effective dose for computed tomography, CT, examinations can approximately be derived from the DLP value, either by applying a examination type specific conversion factor or by using a Monte Carlo based computer program that take into consideration each slice. In this project dose values calculated by conversion factors provided by SSI (FS 2002:2) were compared to those calculated by ImPACT (ver. 0.99r). Both methods are based on simulations done by Jones and Shrimpton (1993).

The calculations of AGD for mammography were divided into three levels. Level 1 was based on the values given by the X-ray equipment at hand while level 2 used AGD=ESAK*p*g suggested by Dance (1990) where p converts the entrance surface air kerma (ESAK) for polymethyl methacrylate (PMMA) to the standard breast (50:50 adipose and glandular tissue) and g converts to AGD. Both parameters depend on beam quality. These two levels were compared to level 3, which also considers the breast glandularity as a function of breast thickness and age, as suggested by Dance et al (2000).

The coefficient of determination in the comparisons of estimation levels to the best estimation possible was calculated as

$$R^2 = 1 - \frac{\sum (E_i - \hat{E}_i)^2}{\sum (E_i - \overline{E})^2} \quad (1)$$

where E_i is the estimation to $\hat{E}_i$ and $\overline{E}$ as the mean value. The mean relative error was calculated as

$$\overline{\varepsilon}_r = \sum \frac{\left| E_i - \hat{E}_i \right|}{\hat{E}_i} / N \quad (2)$$

Results and discussion

The accuracy in the estimations of effective dose increased by added information (projections, exposure data, patient data etc.). Correlation coefficients and mean relative errors are shown in table 1. E.g. the relative error of the urography dose calculations decreased from 29% to 14%. The DICOM headers provide (if used accordingly) information of patient dose, exposure data and some patient data. The patient dimensions can be derived from exposure data at a given AEC setting. If access to PACS is granted, data from all examinations can be extracted and processed.

A comparison of the accuracy in determination of the effective dose based on different amounts of information on how the examination was performed was made. The conversion factors between the effective dose and DAP and DLP, respectively, derived by SSI were useful but were improved when knowledge about the patient gender and size as well as position of the radiation field on the patient was obtained.

Table 1. The coefficient of determination, R^2 (1), to the best estimation and the mean relative error, $\overline{\varepsilon}_r$ (2).

Examination type	Level 1		Level 2		Level 3		Level 4	
	R^2	$\overline{\varepsilon}_r$	R^2	$\overline{\varepsilon}_r$	R^2	$\overline{\varepsilon}_r$	R^2	$\overline{\varepsilon}_r$
Radiography								
Pelvis	0.60	39%	0.50	51%	0.50	58%	0.87	32%
Lung	0.65	26%	0.75	24%	0.78	23%	0.77	22%
Spine	0.27	38%	0.26	29%	0.35	33%	0.65	16%
Urography	0.82	29%	0.75	24%	0.69	26%	0.52	14%
Mammography	0.90	10%	0.95	10%				
CT								
Skull	0.99	2%						
Chest	0.91	6%						
Abdomen	0.79	16%						

Conclusions

Calculating the effective dose for each examination using the stored information in PACS simplifies the dose registration procedure. Conversion factors provided by SSI make it possible to calculate effective dose in a convenient way. The accuracy can be improved by using specific conversion factors based on more detailed information, especially in conventional radiography where the parameters can vary substantially due to variations in methodology and patient size. In mammography and CT the exposure settings are more fixed and the dose estimations gave little, but still, room for improvements.

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Session VIII: Industrial uses of radiation

Industrial radiography in Norway
New regulatory regime for operator certification

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Abstract: Industrial use of sealed radioactive sources in Norway is described in terms of inventory of category 1 and 2 sources, together with more detailed statistics for the industrial radiography industry. This is a high risk area applying high risk sources, and the regulatory instruments applied by the Norwegian radiation protection authority in this area are presented. Particularly the new regulatory regime of operator certification by external accredited personnel certification bodies is described

Introduction.

Industrial use of ionizing radiation often involves the use of large sealed radioactive sources. The International atomic energy agency (IAEA) has provided a categorization system for radioactive sources, IAEA (2003). In this system category 1 represent the most dangerous sources from a health effect point of view, with increasing category number signifying decreasing level of risk. The classification system consists of 5 different categories, and in table 1 the total Norwegian inventory of category 1 and 2 sources is shown.

Table 1. Norwegian inventory of Category 1 & 2 Sources.

	Number of undertakings	Number of sources	IAEA Category
Irradiation facilities	1	1	1
Blood irradiation facilities	9	9	1
Gammaknife	1	1	1
Teletherapy	2	2	1
Industrial radiography	88	198	2
Brachytherapy (afterloading)	4	4	2
Total	105	215	

As seen from the table, the majority of high risk sources in Norway are used within industrial radiography. This is a large industry in Norway, due to the extensive offshore activities. The major radiation sources applied are X-ray machines (typical 200 – 300kV), and radioactive sources of ^{192}Ir (1500 GBq) and ^{75}Se (3000 GBq). To a small degree ^{60}Co sources are also applied, and in special situations even portable accelerators may be used. The latter is well suited for testing of very thick objects, and is generally hired for particular projects. A 6 MeV betatron may be used for concrete objects up to 1 m thickness.

Industrial radiography

In table 2 some statistics from the Norwegian industrial radiography industry is shown.

Table 2. Statistics from Norwegian industrial radiography, as of April 7th 2005.

	Number
Industrial radiography licenses	88
Radiation protection certified operators	518
Operators without radiation protection certification	31
Cameras for gamma radiography	198
X-ray machines	187
Companies having one or more shielded enclosure	42
Total number of closed shielded enclosures	63

Site radiography, i.e. radiography outside shielded enclosures, is usually done with radioactive sources. As opposed to X-ray machines, radiation sources do not apply electric power, and is thus the preferred option in the oil industry. Site radiography with radioactive sources has a number of high risk features: Large gamma sources are placed in unshielded positions for radiography in non-standardized environments like factories and mechanical work shops etc. During radiography the sources are brought in and out of shielded positions, with a risk of the source getting jammed, unintentional brought into unshielded position, lost etc. The dose rate from an unshielded radiography source is typical 100 – 200 mSv/h at 1 m distance. In Norway there is annually 5 – 10 reported incidents/minor accidents involving radiography sources, and globally there is frequent reports of serious accidents, some even fatal, with radiography sources. Data over accidents with clinical consequences to occupationally exposed workers for the time period 1975 -2000 have been reported by Unscear (2000). In table 3 Unscear data are presented in cumulated form.

Table 3. Accidents with clinical consequences to occupationally exposed workers 1975-2000.

Field of application	Number of accidents
Nuclear fuel cycle	9
Industrial use of radiation in total	65
Industrial radiography	36
Tertiary education and non industrial use	13
Medical uses of radiation	12
Total number	99

Due to the high risk nature of NDT work, there are strict regulatory requirements for this industry. Companies performing industrial radiography must be authorized by the Norwegian radiation protection authority (NRPA). One of the requirements in the authorization is that site radiography must be performed with 2 certified operators, while for radiography in shielded enclosures one certified operator is required. Previously the radiation protection certification of operators was done by the NRPA. From January 1st 2005 this function is entirely delegated to external, independent, accredited certification bodies. The certification body has to be accredited for this particular personnel certification function. A Normative document, describing the necessary radiation protection training and qualifications of industrial radiography operators, is one of the basic elements in the accreditation and certification process. Such a Normative document has been prepared in a cooperation between NRPA and the Norwegian society for non-destructive testing, and is issued as a formal NRPA document as StrålevernHefte 28. The document may be downloaded from the NRPA web site. The other basic document in the accreditation process is European and international standard EN ISO/IEC 17024 (2003).

Training requirements and radiation protection certification.

In the Normative document, competence within the following topics is required for radiation protection certification in industrial radiography:

Regulatory requirements for using radiation sources for industrial radiography.

Basic properties of X-and gamma radiation.

Radiation output of the various kinds of radiation sources, and be able to use this information for dose calculations.

Hazards and health effects from ionizing radiation.

Radiation protection terminology

Regulations for transport of radioactive sources, to a level corresponding to the specialized course for class 7.

Basic principles for radiation protection, and be able to performed calculations with time, distance and shielding.

Use of radiation protection instrumentation.

Practical means for operational dose reduction.

Correct work procedures with regard to measurements of radiation levels, erecting barriers, daily control and use of sources and equipment.

Service, maintenance and calibration of equipment.

Be able to detect abnormal situations, and to make the proper corrective actions.

Reconstruction and dose estimations in connections with accidents.

Writing reports and keeping logbooks

The Normative document also describes the content of a standard 35 hours training course regarded necessary for obtaining the qualifications listed above, together with a

description of the examination process and the minimum requirements for passing the examination. In particular cases, Norwegian radiation protection certificates may be issued on the basis of foreign certificates or training.

For the time being there are two certification bodies performing this radiation protection certification in Norway. One of the certification bodies is based in Sweden, and is being accredited by Swedac. The other is Norwegian, and is being accredited by Norsk Akkreditering. The new certification regime thus established implies that the radiation protection qualifications of operators are kept under surveillance by the national accreditation bodies, and not by the radiation protection authorities.

Conclusive Remarks

The majority of international reported serious accidents within industrial radiography are caused by operator errors or failures to follow procedures, while a minor part is caused by equipment failure alone. The equipment used in Norway for gamma radiography is the same as used elsewhere in the world and the same as equipment used in a number of reported serious accidents abroad. Training of operators and establishing of safety culture has been of vital importance for improving the radiation safety in industrial radiography, and we believe that the established system of operator certification is a useful tool in this respect. We also think that transferring the certification process to accredited external bodies will ensure a better follow up and maintenance of competence and training of the certificate holders, according to the accepted international principles described in EN ISO/IEC 17024.

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Supervision of radiography licensees -using electronic questionnaires

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Supervision of radiography licensees

-using electronic questionnaires

Å. Olson

In Sweden there are about 140 companies with a license for radiography, including both site radiography and radiography in closed areas.

In October 2003 the Swedish Radiation Protection Authority, SSI, decided to send out an electronic questionnaire in order to investigate to what amount the Swedish regulations on radiography, SSI FS 2000:8, are fulfilled, and also to increase the knowledge of these regulations.

The questionnaire

The questionnaire is in Excel format, containing one part with general questions about the company and their practice, and one part where the company should indicate to that degree they fulfil the different paragraphs in the SSI regulations SSI FS 2000:8. The electronic format makes it easy to collect, process, and analyse the data.

Response

In all, 93 out of 140 completed questionnaires were returned to SSI. A number of licensees had different technical problems with the questionnaire, which resulted in 21 answers on paper.

Results

22% of the answering companies are working only with site radiography, 73% only with radiography in closed areas, and 5% with both types of radiography.

86% use only x-ray equipment, 2% use only radioactive sources, and 12% use both types.

The results show that the licensees have most difficulties with the following paragraphs:

- § 9:12, routines for keeping journals, reporting and keeping of operational statistics.
- § 3, radiation protection organisation described in an organisation scheme.
- § 30, a closed area shall be equipped with warning symbol, information about the type of radiation source, information about where the manual can be found, and the name of the contact person.

Part 1:General information

Statens strålskyddsinstitut

Questions part 1.

1. When was the last time your practice was inspected by SSI?

2. Is your practice to be considered as
   - ☐ Site radiography
   - ☐ Radiography in closed areas

3. Is your practice carried out with
   - ☐ X-ray equipment
   - ☐ Radioactive source

4. Is your radiography practice performed at more than one location within the same company?
   - ☐ Yes
   - ☐ No

5. How many irradiations do you perform each year?
   - ☐ 0-9
   - ☐ 10-99
   - ☐ 100-500
   - ☐ >500
   - ☐ The equipment is at a conveyor belt or similar

Example of part 2: The regulations

Statens strålskyddsinstitut

Questions part 2.

SSI FS 2000:8

The questions below applies to the SSI regulations on radiography, SSI FS 2000:8. In the column to the right you can mark the alternative that best represents the conditions in your workplace.

Estimate to what degree your company meets the demands in each paragraph

1=Not at all
2=Partly
3=Good
?=I don't understand the question
Not relevant= Not relevant for our practice

Please note that 7§, 8§, 9§ first sub-clause, and 14-27 §§ is only applicable to site radiography (def. see 7§)

	1 2 3 ? Not relevant
Radiation Protection Organisation § 3 The licence holder shall have a radiation protection organisation which is adjusted with respect to the extension and kind of the practice and is described in an organisation scheme. The organisation scheme shall show how the tasks are distributed between the persons with respect to radiation protection matters. The organisation scheme shall be kept up to date.	○ ○ ○ ○ ○
Competence General § 4 The licence holder shall ensure that all workers who take part in the practice have the theoretical and practical knowledge needed to perform the work in an sound way from a radiation protection point of view.	○ ○ ○ ○ ○

Summary

The results from this questionnaire gave a general view of the state of the radiography practices in Sweden, and the questionnaire hopefully contributed to improve the knowledge of these regulations for the licensees.

The largest deficiencies were found in the documentations of the radiation protection organisation as well as documentation of quality assurance and operational statistics.

The results of the questionnaire constitute a useful basis for selecting companies for future inspections by SSI. Those who has not answered the questionnaire and are working with site radiography can expect a visit in the near future…

Session IX: Radiobiology

Radiation biology at the cellular level, basic consideration with focus on the consequences of low doses and dose rates

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Ionising radiation induces chemical alterations in all cellular components. In the nano scale radiochemistry is determined by the quality of the radiation and the properties of target. DNA, the primary target of biological concern, is organised in compartments where the radiation chemistry is largely influenced by micro structures leading to different complexity of the primary damage. The quantity and the quality of the primary damage will determine the biological response, a scenario with many possible choices and pathways, with the aim to minimise the negative consequences. The primary responses such as induction of repair pathways, cell cycle control and cell death will be reviewed.

The primary question in the context of risk estimates for low dose and dose rates is the shape of the dose response curve for stochastic effects.

The development of new techniques have rapidly increased our knowledge regarding the mechanisms initiated by the primary damage and the subsequent cellular responses thus the possibilities to analyse consequences of dose rates in the range of a few mSv/year are coming closer. Recent data with implications on dose response relations for genotoxic effects of doses inducing less then one hit per cell will be discussed.

Non-targeted effects of ionising radiation - A challenge to the current radiobiological paradigm

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A basic paradigm in radiobiology is that, after exposure to ionising radiation, the deposition of energy in the cell nucleus and the resulting damage to DNA, the primary target, are responsible for the harmful biological effects of radiation. The radiation-induced changes are thought to be fixed already in the first cell division following the radiation exposure and health effects are considered to result as a consequence of clonal proliferation of cells carrying mutations in specific genes. Since the initial damage induced in DNA has been shown to be directly proportional to dose, risk is also be considered to be directly proportional to dose. Risk from multiple exposures is considered to be additive, and risk from high and low LET radiation exposure is assumed to be qualitatively the same. These assumptions are incorporated into the Linear-No-Threshold (LNT) Hypothesis that is used in all radiation protection practices.

A range of evidence has now emerged that challenges the universality of the target theory of radiation induced effects. These effects have also been termed "non-(DNA)-targeted" and include radiation-induced bystander effects, genomic instability, adaptive response, low-dose hyper-radiosensitivity, abscopal (out-of-field) effects of radiotherapy, clastogenic factors, delayed reproductive death and induction of genes by radiation. Essential features of non-targeted effects are that they do not require direct nuclear exposure by radiation and they are particularly significant at low doses. This new evidence suggests a need for a *new paradigm* in radiation biology. The new paradigm should cover both the classical (targeted) and the non-targeted effects. New aspects include the role of cellular communication and tissue-level responses.

A better understanding of non-targeted effects may have important consequences on the health risk assessment and, consequently, on radiation protection. The non-targeted effects may contribute to the estimation of cancer risk from occupational, medical and environmental exposures. In particular, they may have implications for the applicability of the Linear-No-Threshold (LNT) model in extrapolating radiation risk data into the low-dose region. This also means that the adequacy of the concept of dose to estimate risk is challenged by these findings. Moreover, these effects may provide new mechanistic explanations for the development of non-cancer diseases. Further research is required to determine if these effects, typically measured in cell cultures, are applicable in tissue level, whole animals, and ultimately in humans.

Genomic Instability

Radiation-induced genomic instability means that the progeny of irradiated cells show occurrence of new mutations and/or new chromosomal aberrations or other genomic damage for many generations. Affected progeny also demonstrate high levels of lethal mutation, which may be measured as delayed reproductive cell death and/or delayed

apoptosis. These effects occur in cells that were not exposed to radiation. Genomic instability occurs in the progeny of irradiated cells at a frequency that is several orders of magnitude higher than would be expected for a mutation of a specific gene. Therefore a mutation in, for example, a repair gene is not a likely explanation for the induction of genomic instability. Genomic instability is induced both by high-LET and low-LET radiation, but not all cell lines show this effect. The dose-effect relationship for genomic instability invariably shows a plateau but is a function of time at which effects are scored. High LET is more effective than low LET, but LET also influences the temporal pattern of expression.

Animal studies indicate that some mouse strains are genetically more susceptible to genomic instability induction than others. These strains also show a higher susceptibility to radiation-induced malignancy. Genotypes that have a less effective apoptotic response seem to be more predisposed to the development of malignancy.

Bystander effects

Bystander effects are changes in cells that were not directly hit by radiation but were nearby. The signal can be transferred via the culture medium (‘clastogenic factors’) or cell-to-cell communication (inhibition of cell communication prevents bystander effects). Bystander effects have been described in a variety of cellular systems and in tissue explants.

Bystander effects are not new. Starting from the 1960’s, there is extensive literature on clastogenic factors and other ‘compounds’ that stimulate or modify responses in cells that were not damaged. Modern microbeam exposure systems capable of exposing single cells or even defined cellular organelles to charged particles or ultra soft X-rays have facilitated research on bystander effects. Such irradiation facilities also make it possible to target subcellular structures, such as nucleus, cytoplasm or mitochondria with either a single or an exact number of alpha particles. The dose-effect relationship for bystander effect invariably shows a plateau below one gray. Moreover, the effect appears to be determined by dose per hit cell, rather than number of cells hit, and high and low LET radiations appear to be equally effective. Bystander effects are the most likely drivers for the more delayed non-targeted effects such as genomic instability and adaptive response.

Individual sensitivity seems to play a role both in genomic instability and bystander effect. Genotypes that have a more effective apoptotic response seem to be less predisposed to the development of malignancy. The genetic basis for this variability requires further research.

Adaptive response

Adaptive response is a biological phenomenon in which resistance to a challenging dose of radiation is established by one or several very small preceding doses. Therefore, adaptive response may be an important modifier of risk in situations where radiation exposure is protracted. Generally, and unlike most data available for genomic instability and bystander effect, the adaptive response depends on synthesis of proteins, most of which are involved in DNA damage response.

Very few studies so far have tried to investigate the relationships between bystander effect, genomic instability and adaptive response. However, it has been shown that the induction of adaptive response by low-LET radiation can protect against bystander damage induced by alpha particles. Moreover, these studies on cell cultures have shown that all three effects (genomic instability, bystander effect and adaptive response) may be observed at time points distant from the initial radiation exposure. These results extend the adaptive response to include environmentally relevant exposure situations, ie. where the challenging dose may be far removed from an initial dose, and may affect cells that were not themselves originally irradiated.

Also UV and chemicals induce non-targeted effects

Induction of genomic instability or bystander effects is not unique to ionising radiation only, but it is known that also UV and genotoxic chemicals can cause such effects. Ionising radiation is, however, a good model for studying delayed damage, because no extra substance is left in the cells after external irradiation. In case of chemicals, interpretation of delayed damage is more complicated because of the possibility that traces of chemical may remain in the cells and still cause effects in subsequent cell generations.

The fact that several environmental agents and contaminants are capable of inducing bystander effects and genomic instability, suggests that bystander signalling and related effects may be related to a more universal response system to external stimuli/insults.

Low dose effects

The cancer risk at low doses will probably never be fully elucidated by epidemiological studies, as this would require very large populations and accurate dosimetry. The dosimetry of protracted exposures is even more demanding than dosimetry for single exposures. Uncertainties in dosimetry of epidemiological studies make it more difficult to observe a dose response, which in turn tends to lead to lower risk estimates. Biological modelling of radiation carcinogenesis may offer a tool to study the risk at the low dose region. The input data should contain not only the conventional direct radiation effects but also non-targeted effects which may be important modifiers of risk at the low dose region. It remains to be determined how this would apply to low-level radiation and whether it would increase, decrease, or leave unaltered, current assessments of risk.

Genomic instability and bystander effects are observed already after very low doses. Using a microbeam exposure system, it has been shown that a single alpha particle is able to induce chromosomal instability in the progeny of cultured human cells. In fact, the dose response data indicate that the relative contribution of these indirect effects as compared to damage caused by direct hits may well be more pronounced in the low dose region, thus giving some support for a potential supralinear response in the low-dose region.

The genomic instability and bystander endpoints are both transmissible (mutational) and non-transmissible (lethal). The balance of these in different cellular systems may lead either to an increased or decreased risk. Some scientists indeed argue that these non-targeted radiation effects are in fact part of the adaptive response to ionising radiation and therefore protective. More research is needed on the delayed damage response systems, such as adaptive response and premature differentiation. An increase in cancer risk can

be argued by amplified genomic damage, genomic instability and also by increased proliferation of cells due to cell killing. A decrease in cancer risk can be argued by cell killing removing damaged cells and adaptive response and increased differentiation of cells which may protect. During embryonic and foetal development, however, any changes altering the normal pattern of cell proliferation, cell differentiation and cell migration are likely to be harmful.

Implications for radiation protection

The main source of information on radiation-induced human cancer risk comes from epidemiological data on exposed populations. Direct information is available only at relatively large doses, and linear extrapolation from this data is applied at lower doses, which are more relevant in terms of exposure to the general population and radiation workers. The shape of the dose response curve for cancer at low doses is a matter of constant debate. Arguments range from a threshold or even beneficial effect of small radiation doses (hormesis) to non-threshold supralinear responses (implying that small doses are more hazardous than previously assumed). Genomic instability, bystander effect and adaptive response provide new insight in our understanding of low dose effects.

Further research on non-targeted effects may eventually contribute to the formulation of a new radiation biology paradigm combining both the classical (targeted or direct) and the new evidence on non-targeted (indirect) radiation effects. This may have profound implications for risk assessment of ionising radiation.

Non-cancer effects

Some data on radiation-induced genomic instability suggests that perpetuation involves epigenetic mechanisms. Increased oxidative stress seems to be a long term characteristic of the progeny of irradiated cells and animal studies have suggested inflammatory-type responses. Non-targeted effects may provide a potential mechanistic explanation for the development of non-cancer diseases. The well-documented increases in malignancy in the Japanese a-bomb survivors have recently been supplemented by reports of increases in cardiovascular, digestive and respiratory system diseases. Such effects are very difficult to explain on the basis of the conventional target theory, but could be linked to oxidative stress and inflammatory-type of responses.

Since non-targeted cellular responses to radiation are the products of cell signalling which result in modulation of a variety of genes, including those that produce free radical scavengers and enzymes to repair DNA damage, it is expected that such exposures could impact on the risk of non-cancer effects as well as on the risk of cancer. Research to date indicates that both of these cellular responses show an “all or nothing” type of response to dose, suggesting that the first track of radiation produces the maximum gene response. If this is so, then the radiation protection concept of an effect that is proportional to dose is inaccurate at low doses, and this difficulty may apply equally to non-cancer and cancer endpoints.

Studies of non-targeted effects of ionizing radiation

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Abstract: Non-targeted effects of ionising radiation defined as phenomena, which do not require direct DNA damage and include bystander effect, genomic instability and adaptive response. Recent evidence demonstrated that they might be interrelated and possibly have a protective role under *in vivo* conditions. Our own studies of bystander effect and genomic instability in telomerase immortalised normal human cells and 3D artificial tissue models demonstrate a strong link between bystander effect and genomic instability. Our findings also suggest that these effects might relate to adaptive response because of increased non-targeted differentiation in irradiated samples. Based on these experimental data we propose a theory that the main function of the non-targeted effects is to decrease the risk of carcinogenesis in a multicellular organism exposed to oxidative damage (including radiation induced). It can be speculated that an integrated multicellular system may be able to detect oxidative damage and respond to it by removing a functional group of cells, which could be potentially damaged. Such a mechanism of cooperative response would make the tissue system much more robust.

Introduction

The radiation-induced bystander effect is a phenomenon whereby cellular effects such as sister chromatid exchanges Nagasawa and Little (1992), chromosome aberrations Watson *et al.* (2000), apoptosis, micronucleation Belyakov *et al.* (2001 and 2002), transformation Sawant *et al.* (2001), mutations Zhou *et al.* (2001) and changes of gene expression Azzam *et al.* (2001) are expressed in *unirradiated neighbouring* cells near to an irradiated cell or cells. Radiation-induced genomic instability is defined as a persistent elevation in the rate of *de novo* appearance of genetic changes such as mutations, chromosome aberrations or micronuclei Belyakov *et al.* (1999) within a clonal population Wright (2000). Genomic instability and the bystander effect are both non-targeted effects of irradiation. They have a cross-section much larger than the nucleus. The bystander effect and genomic instability might be related phenomena. It was reported that persistent genomic instability can be induced via a bystander mechanism under *in vitro* Lorimore *et al.* (1998) and *in vivo* Watson *et al.* (2000) conditions. This evidence suggests that the cells that are affected by the bystander mechanism may remain at an increased risk of genetic alterations for many generations. Other studies have suggested a common relationship between genomic instability and the bystander response. Some evidence of protective function of bystander effect is available Barcellos-Hoff and Brooks (2001). There is some indication that genomic instability may play a protective role as well. It was recently demonstrated Limoli *et al.* (2001) that chromosome instability can lead to the development of cell variants that are more resistant to radiation. Bystander effect and genomic instability might be parts of a comprehensive system of oxidative damage control, which aims to reduce the risk of carcinogenesis Pollycove (1998). Finally there are suggestions that both the bystander effect and genomic instability are controlled through epigenetic mechanisms such as DNA methylation Trosko (1998).

Material and methods

Our work relates to experimentation with infinity telomerase-immortalized cell lines (human fibroblasts hTERT-BJ1 and human retinal pigment epithelial cells hTERT-RPE1), which allow investigation of bystander effect and genomic instability in wide time frame. However, the non-targeted effects cannot be comprehensively explained on the basis of a single cell reaction. It is well known that an organism is composed of different cell types that interact as functional units in a way to maintain normal tissue function. Radiation effects at the tissue level under normal conditions prove that individual cells cannot be considered as an isolated functional unit within most tissues of a multicellular organism. Therefore the radiation response is not simply the sum of cellular responses as assumed in classical radiobiology, predominantly from studies using cell cultures. Experimental models, which maintain tissue-like intercellular cell signalling and three-dimensional (3D) structure, are essential for proper understanding of bystander effect and genomic instability. The tissue microenvironment is also important for proper manifestation of non-targeted effects. Extracellular signalling in normal tissues plays a crucial role in initiation and perpetuation of both bystander effect and genomic instability. In order to fulfil these requirements we studied radiation-induced bystander effects and genomic instability in novel 3D artificial human skin tissue systems available from MatTek corporation (Boston, USA). Air-liquid interface culture technique is used to grow artificial tissues, which allow to model conditions present *in vivo*.

To treat cell culture samples, we were using “medium transfer technique” developed by Mothersill and Seymour (1997). “Donor” cultures were irradiated with broad field X-rays then medium from these flasks was filtered and added to the “recipient” samples according to a standard protocol. The Gray Cancer Institute (Northwood, UK) charged particle microbeam was used to irradiate tissue samples in a known pattern with a known number of $^{3}He^{2+}$ particles or protons. After irradiation, the tissues models were incubated for 3 days, fixed in 10% NBF, paraffin embedded and then sliced into 5 μm histological sections located at varying distances from the plane of the irradiated cells. In both cases of cell cultures and tissues we studied cell survival, micronucleation, *in situ* apoptosis and markers of differentiation.

Results

Briefly, we have demonstrated a clear bystander increase in micronucleation and apoptosis for both cell and tissue systems, studied the role of differentiation versus damage induction processes. Evidence of interaction between bystander response and genomic instability was shown in telomerase-immortalized cell lines.

We observed 5-10% reduction of cell survival, 3-5 % increase of micronucleation and 1-3 % increase of apoptosis in infinity telomerase-immortalized cell lines after treatment with medium from irradiated cells at different timepoints (even tens of generation after the initial treatment). Significantly elevated bystander induced apoptosis was observed with 3’-OH DNA end-labelling based technique in 3D artificial tissue skin systems. Pilot data on the bystander effect after microbeam irradiation (measured as increased fraction of damaged cells and bystander induced differentiation) were obtained. Our results suggested an importance of proliferation and differentiation status for bystander effect induction. A single 2 μm location on tissue section was pre-irradiated with 1-10 $^{3}He^{2+}$ particles (5 MeV; LET 75 keV/μm) using microbeam system. Studies of bystander-induced differentiation under *in situ* were performed using morphological measurements in underdeveloped EPI-201 model. Even although only a single region of the tissue

section was targeted, thousands of additional cells were found to undergo bystander induced differentiation. This resulted in an overall increase in the fraction of differentiated cells for approximately 10-15%, which are much greater than that observed for the induction of damage (not more than 1-2% of apoptotic cells).

Discussion

The discovery of ionising radiation induced non-targeted effects is important for understanding the dose-response mechanisms relevant to low-dose irradiation *in vivo*. One important question is whether the non-targeted effects relates to a protective mechanism or whether, conversely, it amplifies the number of cells damaged by the isolated radiation tracks of low-dose exposures leading to an increased risk of carcinogenesis. One theory, supported by the experimental data obtained during this project is that the main functions of the non-targeted effects are to decrease the risk of transformation in a multicellular organism exposed to radiation. It can be speculated that individual cells within a tissue may not have the ability to detect irradiation such that an individual cell response is not expressed. An integrated multicellular system may be able to detect damage from irradiation and respond to it by removing a functional group of cells, which could be potentially damaged. However, not every cell will respond to the hypothetical non-targeted factor(s), which are released by targeted (or instable) cells. Only 1-3% of the total number of cells in the system would express lethal response (micronucleation or apoptosis) and approximately 10-15% would go on to premature induced differentiation. Differences in the gene expression profiles, temporal and spatial patterns of key proteins expressed in directly irradiated and bystander cells may determine how the cells ultimately respond to low doses of radiation. Such a mechanism of co-operative response would make the tissue system much more robust. It would work only for low doses of charged particle irradiation (below~0.1-0.2 Gy, depending on system and type of radiation) because only in this case is the damage localised within a small fraction of the cell population. In some systems, the most convenient way to remove potentially damaged cells is via apoptosis. In particular, apoptosis allows the removal of affected cells without a negative impact on other cells via inflammatory responses. However many apoptotic pathways are controlled by cellular signals, which would also enable the selective removal of certain functional groups of cells. Another way to isolate damage is to prompt affected cells into irreversible differentiation. Underlying this theory is that normal 3D tissue microarchitecture is essential for the manifestation of the effect. Therefore, protective function of non-targeted effects might be a tissue-specific epigenetic phenomenon, which can be observed in full scale when there is presence of natural cellular stratification with differentiated and dividing cells present and an intact tissue microenvironment. However, the data suggest that initial nuclear damage seems to be essential for initiation of this system. Perpetuation of the bystander effect genomic instability might involve cascade-like epigenetic mechanisms. Tissues remove all potentially damaged cells from the system to avoid the risk of carcinogensis following sparse low dose irradiation or any other local oxidative damage. Bystander induced differentiation seems to play a central role in this process, see also Belyakov *et al.* (2002 and 2005).

Acknowledgments

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Present status of the studies on biological effects of ionising radiation in Norway

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Institutions with activities in the field

- **The Norwegian Radium Hospital,** ***Institute of Cancer Research,***
- **University of Oslo,** ***Dept of Physics, The Biophysics group***
- **Norwegian University of Science and Technology**
- **National Institute of Public Health,** ***Dept of Chemical Toxicology***
- **The Norwegian University of Life Sciences,** ***Dept of Plant and Environmental Research***
- **Ullevaal University Hospital** ***Dept of Oncology***
- ***Cancer Registry of Norway***
- **Norwegian Radiation Protection Authority**

Norwegian Radiation Protection Authority

Biological effects of ionizing radiation and CANCER

the Norwegian Radium Hospital

Institute for Cancer Research, Department of Radiation Biology

Radiation biology & tumour physiology (RBTP)

Molecular Radiation Biology

Experimental Radiation Therapy & BioART

Norwegian Radiation Protection Authority

Biological effects of ionizing radiation and CANCER

Main activities at the Radium Hospital group:

Tumour response to radiation

Molecular mechanisms of radiation response of cancer in cervical carcinomas and malignant lymphomas

Tumour physiology and radiation response of cancer

Impact of hypoxia in radiation response.

Normal tissue response to radiation

Genomic instability and cancer

Late normal tissue sequela

Bio-ART (biological adaptive radiation therapy)

Norwegian Radiation Protection Authority

Biological effects of ionizing radiation and CANCER

Other groups

- ***Ullevål University hospital*** **- experimental studies on radiation enteropathy, molecular mechanisms as well as modeling in USA in collaboration with Prof. Martin Hauer- Jensen at Arkansas Cancer Research Center**

  St. Olavs Hospital in Trondheim and NTNU **Institute for Cancer Research and Molecular Medicine - studies on molecular aspects of PDT**
- ***Dept.of Oncology, St. Olavs Hospital*** **- research on radiation sensitivity of breast cancer cells in culture very often and interaction with drugs and growth factors**

Norwegian Radiation Protection Authority

Projects within radiobiology
University of Oslo Dept of Physics

Biophysics and Medical Physics Section

Studies of low-dose hyperradiosensitivity

Long-term hypoxia. Extreme hypoxia- role of the retinoblastoma protein in regulatory responses during hypoxia and low-dose rate irradiation.

ESR studies of energy transfer between DNA and histones

Norwegian Radiation Protection Authority

Radiation, DNA damage and repair

Focus on oxidative damage

The Norwegian Institute of Public Health, Department of Chemical Toxicology

- **Repair of oxidative DNA lesions in male germ cells (rodents and humans)**
- **Effects of ionising radiation from radiocesium in soil on earthworms. Effects of 137Cs in soil are studied in the** *Eisenia fetida*

The National Hospital (Rikshospitalet)

- **work on oxidative DNA damage and repair but also on ionizing radiation i.e. studies on the role of Flap endonuclease 1(FEN1) -**

Norwegian Radiation Protection Authority

Ecological studies of biological effects of ionizing radiation

New focus on studies Norway on low dose rate chronic irradiation (UMB, NRPA, NIPH)

Several years ago NRPA initiated the study on biological effects of chronic exposure to radionuclides.

Influence of chronic exposure to Cs137 on reproduction system in earthworms. Significant efforts were made to study effects on spermatogenesis (main experimental work at UMB in collaboration with NIPH)

Norwegian Radiation Protection Authority

Ecological studies of biological effects of radiation

Facility at The Norwegian University of Life Sciences

Figure 1. The gamma radiation source. The ^{137}Cs source recently established at NLH, is unique, providing transitory to low dose rates, allowing long-term low dose chronic exposure experiments to be performed. The exposure field (experimental room, 2.9 x 10 m) is well calibrated; high dose rate (~ 1.4 Gy/h) 20 cm from the source, decreasing with distance to low dose rate (~ 150 µGy/h) 21 m from the source. Sets of organisms (replicate samples) can be exposed to different dose rates, simultaneously.

Norwegian Radiation Protection Authority

Approach to radiation biology in Norway- some characteristics

- **Do not exist as a separate research field**
- **A part of biophysics education and research**
- **A part of cancer research**
- **Groups working with different types of radiations i.e. ionizing versus non-ionizing not always have enough contact**
- **More focus on scientific applications than filling the knowledge gaps**
- **Very limited teaching under toxicology education programs and medical education programs**
- **Recently more focus on biological effects studies in radioecology**

Norwegian Radiation Protection Authority

Questions

- **Is it sufficient that all radiobiological research in the country is focused on cancer?**
- **Is it possible in such a situation to take care of the education and general knowledge about radiation effects in society?**
- **Will there be a capability in the future to take care of the national risk assessment, for example, in the case of low doses?**

Norwegian Radiation Protection Authority

Session X: Natural sources of radiation

Number of lung cancer deaths caused by radon in Finland

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Abstract: The number of lung cancer cases attributable to indoor radon in homes in Finland was calculated using a risk estimate from a recent European collaborative study and applied to the Finnish context with national demographic, smoking and radon exposure data. The risk model was simplified from the BEIR VI model, with constant excess relative risk per radon exposure for both sexes, and across different age groups and exposure periods. The estimated annual number of lung cancer deaths attributable to indoor radon was 354. Several factors affecting the uncertainty of the count estimate obtained are discussed.

Introduction

Indoor radon exposure received in homes is a well known cause of lung cancer. The European collaborative study published recently yielded more precise risk estimates than earlier studies, with elaborate control of smoking effects (Darby et al. 2005). This motivated us to assess the annual number of cancer deaths attributable to radon in Finland, and to consider the potential cancer deaths that could be averted by radon mitigation. The current reference levels for indoor radon are 400 Bq/m^3 in existing buildings and 200 Bq/m^3 in new buildings (design level).

We aimed at estimating the number of lung cancer deaths attributable to radon using previously developed methodology. The method is based on a risk model with national adaptation based on demographic, smoking and indoor radon data. This approach also enables us to investigate the effect of the model parameters on the deaths averted and, furthermore, on the potential efficacy of the radon policy.

Materials and methods

The BEIR VI age-duration risk model is based on pooled analysis of 11 underground miner cohort studies (National Research Council 1999). The basic form of the excess relative risk is

$$ERR = \beta(\theta_{5\text{-}14}w_{5\text{-}14} + \theta_{15\text{-}24}w_{15\text{-}24}+\theta_{25+}w_{25+})\varphi_{age}\gamma_{dur},$$

where β is excess relative risk per radon exposure, w_i is radon exposure received in time window i before current age and θ_i is a corresponding weighting coefficient. The coefficient φ_{age} refers to attained age, and γ_{dur} to exposure duration.

We used the EC Radon Software (ECRS) to calculate lifetime risks for individuals of different attained age, and with different radon and tobacco exposures (Copyright © 1999-2000 European Commission - DG XI). The ECRS uses databases specifying the risk model parameters, demographic data and smoking statistics.

We applied the proportionate increase of risk of 8.4% per 100 Bq/m^3 of measured radon received during 30 years, obtained in the European collaborative study. In units used in miner studies, this corresponds to the value of 0.0064 WLM^{-1} for the coefficient β, which is slightly larger than 0.0055 WLM^{-1} used in the BEIR VI report. Unlike the BEIR VI report, we chose unity value for all weighting coefficients θ_i, φ_{age} and γ_{dur}. Because the model parameters are not independent, the β-values of these two models are therefore not directly comparable.

Parameters of the used smoking risk model were calculated using risks from the European Collaborative study by weighting the risks in each smoking category (<15, 15-24 and 25+ cigarettes per day for current smokers) by corresponding smoking rates in the year 1985 in Finland, and by rates in the European Collaborative study (ex-smokers <10 and 10+ years) (Darby et al. in press, Piha et al. 1986). Smoking data were from the year 1985, because those were regarded as most relevant for current lung cancer deaths. The lung cancer risk relative to life-long non-smokers for current smokers and ex-smokers was 25 and 12 for males, and 8 and 3 for females, respectively. Latency period of 10 years was assumed. The joint effect of radon and smoking was presumed to be multiplicative.

In our study we used all cause mortality from the year 2003 by 5-year intervals provided by Statistics Finland and lung cancer mortality data provided by the Finnish Cancer Registry. Risk calculations were performed for male and female lifetime non-smokers, smokers starting at the age of 15 years and ex-smokers quitting smoking at the age of 40. Indoor radon exposure (Table 1.) was based on a representative study with a random sample of households performed in 1990 in Finland (Arvela et al. 1993). Stratum-specific means of four indoor radon categories were used in the calculation.

Table 1. Distribution of indoor radon concentration in homes in Finland.

Radon concentration Bq/m^3	% of population	Category Mean Bq/m^3
0-99	63	58
100-199	23	138
200-399	10	274
400 -	4	842

Using ECRS, we calculated lifetime lung cancer death risks for individuals with attained ages from 3 to 88 years with 5 year intervals (18 ages), both genders, 4 radon exposure and 3 smoking categories. We calculated separately the risk without any exposure, that caused by smoking only and that caused by radon only. The average lifetime risk for lung cancer death for all the 2x4x3 = 24 groups were calculated by weighting the risk in each age category by the number of individuals alive in respective age group. Furthermore, the etiologic fraction of lung cancer caused by radon was calculated in each group as the ratio of lifetime lung cancer death risk caused by radon only, relative to that from all causes.

To obtain the number of lung cancer deaths caused by radon, we multiplied the proportion of the Finnish population in the 12 exposure groups of males and females with the number of lung cancer deaths observed by the Finnish Cancer Registry in the year 2003.

Results and discussion

The etiologic fractions of lung cancer cases caused by radon varied from 9.5 % for male smokers in the lowest radon exposure category to 63.2 % for male non-smokers in the highest category (Table 2.). The number of annual lung cancer deaths caused by radon was 354 (Table 3.). This estimate is higher than the value of 200 annual lung cancers used by STUK in its public communication.

Table 2. Etiologic fraction of lung cancer death caused by residential radon in Finland in 2003.

Lifetime indoor radon concentration Bq/m^3	Males			Females		
	Non-smokers	Ex-smokers	Smokers	Non-smokers	Ex-smokers	Smokers
0-99	16.4 %	10.3 %	9.5 %	10.5 %	10.3 %	10.5 %
100-199	22.2 %	20.1 %	19.6 %	19.0 %	20.1 %	21.4 %
200-399	37.7 %	33.6 %	32.1 %	36.0 %	34.3 %	34.6 %
400-	63.2 %	60.1 %	56.5 %	62.9 %	61.2 %	60.4 %

Table 3. Number of lung cancer deaths caused by radon in homes in the year 2003 in Finland.

Lifetime indoor radon concentration (Bq/m^3)	Males	Females	Total
0-99	102	32	134
100-199	69	24	93
200-399	58	23	81
400-	64	28	92
Total	293	107	**400**

Table 3. shows that about 30 % of the lung cancer cases occur among residents living in houses with radon concentration less than 100 Bq/m^3. Only 23 % of the total of 400 estimated lung cancer deaths occurs in the highest radon category, where radon mitigation radon is recommended. Advice to take countermeasures is also given in the category of 200-400 Bq/m^3 where the proportion is 20 %. If we assume a decrease of 70 % in the highest category and 50 % in the second highest, the potential number of avertable cancer deaths would be 105 per year, corresponding to 26 % of all lung cancer deaths due to radon. Applying radon-safe foundation when building new houses would avert more deaths, because it affects all radon levels.

Our model calculation is subject to several uncertainties. The European Collaborative study provided two estimates for excess relative risk: we used 8.4 % per 100 Bq/m^3 calculated directly from the radon measurements. Another ERR estimate of 16 % was ob-

tained based on values corrected for random error, “usual radon”. We chose to use the estimate for measured radon, because the radon exposure distribution was also based on observed, uncorrected values. The use of the estimate for measured radon may underestimate the lung cancer death counts.

The parameter θ_i refers to a dilution effect on radon exposure according to the time since exposure. The European study took into account only the exposure years from 35 to 5 before index date. This means that the use of the same θ_i coefficients for the whole life might overestimate the number of cancer deaths in our study.

The risk estimate of the European study refers to lung cancer incidence, not deaths. However, the mortality in lung cancer is quite close to the incidence. Furthermore, the European study showed also that a larger risk might be related to small cell cancer having better survival rates than other lung cancer types. Moreover, due to radon mitigation and prevention in new buildings, estimates based on radon concentrations from 1990 may overestimate long-term exposure. All these factors are likely to result in overestimation of the risk due to radon.

As to our future plans, we intend to calculate the lung cancer counts using the risk estimate related to “usual radon”. This requires the measured radon distribution to be corrected for the measurement uncertainty.

Our model is based on directly measured etiological factors and observed lung cancer deaths. We aim to develop the model further in order to obtain absolute numbers of lung cancer deaths. Comparing calculated counts to the observed ones will enable us to validate the model and recalculate the estimates using different parameters, if required. The absolute model would also yield separate estimates for smokers and non-smokers, and furthermore, enable us to assess both the effect of decrease in smoking prevalence and radon mitigation on the number of lung cancer cases caused by radon.

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Natural radioactivity in Sweden, exposure to external radiation

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Abstract: Exposure to natural radiation as average gives the main dose to the Swedish population. The individual dose varies largely depending on where people live and their smoking habits. In this paper the authors give an account on the present situation for external exposure and dose. In an accompanying paper we describe the exposure to internal radiation and also the average total dose to the population divided on non-smokers and the whole population.

Introduction

80% of the dose the average Swede receives from ionizing radiation is caused by natural radiation. In Sweden extensive national surveys have been performed and are ongoing in order to investigate the sources of natural radiation and doses to the public and workers. This paper summarizes the present knowledge.

Knowledge of exposure to natural radiation and the range of this exposure is essential for evaluation of risks and need for protection against natural radioactive sources. It also provides a basis for determination of limits for man-made radiation.

Exposure and doses from gamma radiation.

Outdoors: Geophysical mapping of the outdoor gamma radiation started in the early 1950ies in connection with uranium exploration programmes. Since 1968 the Geological Survey of Sweden (SGU) is responsible for these surveys, which include airborne gamma spectrometric measurements, ground surveys and investigation of groundwater resources. Until now 80 % of Sweden has been covered by low altitude airborne surveys at a flight height of 30-50 m and a line spacing of 200 m (Åkerblom, 1993). By a special permission to fly at 50 m over densely populated areas most of these are also included in the surveyed areas, e.g. Stockholm and Gothenburg. At these measurements the activity concentrations of radium-226, thorium-232 and potassium-40 in the ground top surface are recorded as well as the total gamma exposure. The mean gamma radiation at ground level is 90 nGy/h (lakes, rivers and wetlands excluded). As shown in the map Outdoor external exposure in Sweden (Figure 1) the gamma radiation varies strongly between different parts of Sweden. The dominating soil in Sweden is till formed by the land ice during the last glacial epoch when the land ice crushed and grinded the underlying bedrock. The till mainly consists of local rock fragments. For example, as shown in Figure 1, the gamma radiation is low in SE Sweden where limestone, sandstones and gneisses of sedimentary and granodioritic origin occur and in areas in Middle Norrland where greenstones dominate. The areas with the highest average gamma radiation are the areas with Bohus granite at the coast of SW Sweden, areas in Dalecarlia and North Norrland where granite and pegmatite dominates, and alum shale areas in Central and North Sweden Alum shale is a uranium-rich black shale. Depending on where in Sweden it occurs the uranium concentration varies between 50 and 400 mgU/kg (600-5000 Bq ^{238}U/kg)[1] In Table 1 is given data on dominates, and alum shale areas in Central and North Sweden Alum shale is a uranium-rich black shale. Depending on where in Sweden it occurs the uranium

[1] The Swedish limit for indoor gamma radiation in new buildings where people live or work is 0.5 µSv/h. For existing buildings or outdoor exposure no limit is given.

concentration varies between 50 and 400 mgU/kg (600-5000 Bq ^{238}U/kg)[2] In Table 1 is given data on the gamma radiation for different geological formations and the size of these areas.

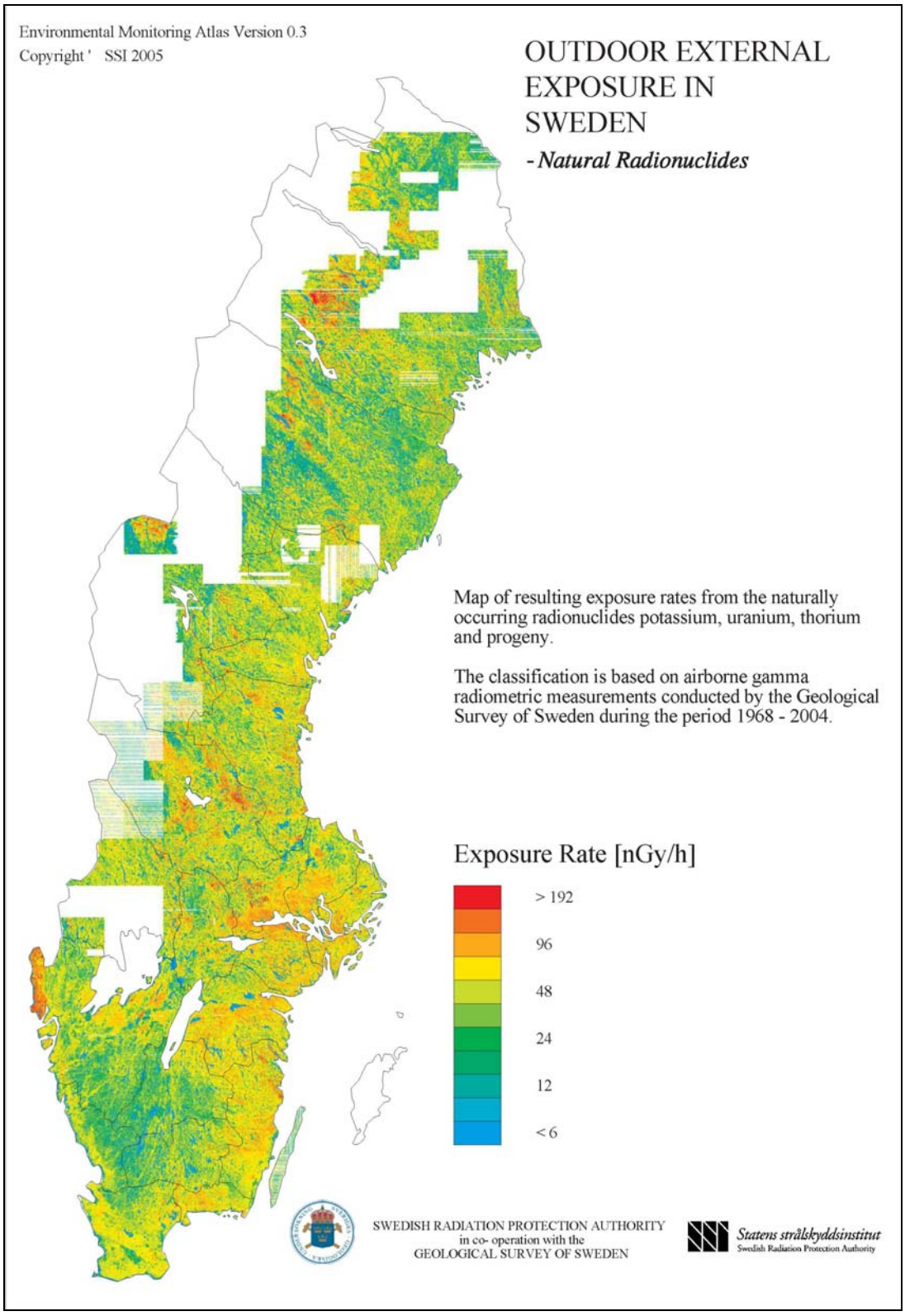

Figure 1. Total exposure to natural gamma radiation at ground level.

[2] The Swedish limit for indoor gamma radiation in new buildings where people live or work is 0.5 µSv/h. For existing buildings or outdoor exposure no limit is given.

The airborne measurements are so detailed that the average exposure form an area of 200 x 200 m can be defined. By matching the data with the demographic register the outdoor individual exposure to Swedes can be determined. Figure 2 shows the outdoor gamma exposure specified for different population groups.

Table 1. External gamma radiation outdoors in Sweden

Normal, nGy/h	Maximum, nGy/h	Source
70		Average exposure from land including wetlands but not lakes and rivers.
15 -130		Normal from exposed bedrock and soils.
170 - 340	500	Areas of granites with enhanced U and Th concentrations.
130 – 850	1700	Areas with alum shale (a uranium-rich black shale).
210 - 1700	1700 - 8500	At > 1000 m^2 large areas of pegmatites rich in U and Th. At smaller sized areas the radiation can be 1700 – 8500 nGy/h.
1700 – 25 000	85 000	At uranium and thorium mineralizations and ores.
	85 000 - 128 500	Measured as maximum from uranium-rich outcrop.

The conversion factor used by UNSCEAR from absorbed dose in air to effective dose, E, is 0.7 Sv/Gy (UNSCEAR 2000). Thus, full time exposure to 170 nGy/h corresponds to a dose of 1 mSv/year. Based on data from the airborne surveys the average yearly dose to Swede's from outdoor radiation is 0.08 mSv/year (assumed average time outdoors, 20 % of the day). However, depending on living habits and location the outdoor dose varies from 0.05 mSv/year on the island of Gotland to 0.2 mSv/year for those living on the Bohus granite (as average approximately 20 % of the year is spent outdoors). For people whose homes are built on exposed alum shale, the dose received from outdoor gamma radiation may reach 1 mSv/year and even higher for those who live on radioactive pegmatites or local uranium and thorium mineralizations.

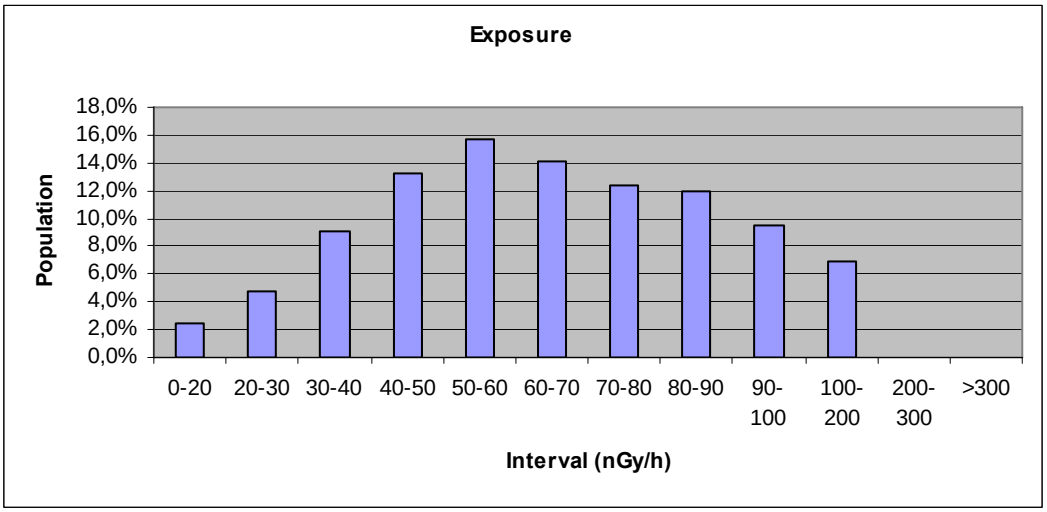

Figure 2. Outdoor gamma exposure specified for different population groups.

Indoors: The indoor exposure to gamma radiation varies from about 20 nGy/h in wooden buildings on ground with low radioactivity to about 1100 nGy/h in buildings with walls, floor and ceilings wholly constructed of lightweight concrete made of alum shale. This material, to larger or lesser extent, has been used for construction of about 400 000 dwellings in Sweden. Even higher gamma radiation, up to 1700 nGy/h has been measured in some few buildings made of slag from radioactive iron ores. In one wooden

house built on bedrock with mineralization of uranium and thorium the indoor gamma radiation was as high as 3000 nGy/h. Most probably, in Sweden you may still find houses built on uranium/thorium deposits in which the gamma radiation can be much higher.

In 1979-1981 SSI investigated the indoor gamma radiation in 1300 dwellings randomly selected all over Sweden (Mjönes, 1981 and 1986) (Table 2). The average indoor exposure was 111 nGy/h. In about 30 000 dwellings the indoor exposure is higher than 425 nGy/h and in approximately 1000 between 800-1100 nGy/h. The use of alum shale-based lightweight concrete ceased 1975-1978. As alum shale-based lightweight concrete is not used any more, the average exposure in dwellings has been reduced since the 1979-1981 investigation, to which level is unknown.

Table 2. Distribution of gamma radiation in Swedish dwellings.

Normal, nGy/h	Maximum, nGy/h	Source
110		Average exposure
17 - 210	3000	In wooden houses (mainly exposure from the ground)
35 - 255	1700	In houses made of bricks and stone materials
250 - 600	1100	In houses of lightweight concrete made of alum shale. About 400,000 dwellings are built of such material

The Swedish population spend approximately 80 % of all time indoors (at home plus at work). Thus, the average dose to the population due to exposure from indoor gamma radiation is 0.54 mSv/h. More than 800,000 Swedes living in dwellings and working in buildings made of alum shale based light weight concrete receive doses of 1.2-1.8 mSv/year, some 60,000 doses of 1.8-2.8 mSv/year and some 6000 3.5-4.3 mSv/year. However, infants and small children spend most of their time indoors and their doses received from gamma radiation is 10-30 % higher than to an adult (UNSCEAR 2000). Thus, for example, a small child living in a house of the most radioactive alum shale-based concrete may receive a dose of almost 7 mSv/year from indoor gamma radiation

Cosmic exposure

Cosmic radiation is produced when particles from outer space, mainly protons, collide with nuclei in the upper atmosphere. These collisions generate a cascade of interactions producing gamma radiation, neutrons, electrons and other elementary particles. The type of radiation produced varies with the altitude. At ground level the dose mainly comes from muons. The size of the dose varies with latitude and altitude and is highest at the Polar Regions and increases with the height above sea level. In Sweden the difference between the southern and northern parts is relatively small. The differences depending on the altitude are small since 80 % of the population lives in areas between sea level and an elevation of 100 m. The average dose to the Swedish population has earlier been estimated to be 0.3 mSv/year. A more accurate recalculation, where population data is taken into account, is planned at SSI. It might result in a slightly higher dose, depending on among other things the increasing number of air flights and the fact that the exposure at flight levels is significantly higher than at ground level.

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Natural radioactivity in Sweden, exposure to internal radiation

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Exposure to radon and thoron

Radon gas in dwellings and workplaces is a large contributor, and in most cases the main contributor, to the doses the Swedish populations receive from natural radiation. The latest national survey of the radon activity concentration in Swedish dwellings was performed 1991-1992 (Swedjemark et al., 1993). Radon concentration was measured in 646 apartments and 714 single-family homes, randomly chosen over all Sweden (Sweden had at that time 2.1 million apartments and 1.9 single-family homes). The mean yearly radon concentration for all dwellings was 108 Bq/m^3, in apartments 75 Bq/m^3 and in single-family homes 141 Bq/m^3 (Table 1)[1]. The main reasons for these relatively high concentrations are the large number of dwellings constructed of alum shale-based lightweight concrete and that the Swedish bedrock and soil which in large parts of the country contains enhanced concentrations of uranium. In Sweden many buildings are found in which the radon concentration exceeds 1000 Bq/m^3 and even 10,000 Bq/m^3. The highest yearly concentration measured in a home was 84,000 Bq/m^3, in a house on an uranium-rich granite. A survey carried out in SW Sweden 2004 in 199 dwellings indicates that the average level now may be about 50 % lower that in 1991-1992 (Ängerheim, 2005) probably due to applied radon preventive constructions in new buildings and successful radon mitigation of radon-affected dwellings.

Table 1. Radon concentrations in Swedish dwellings, 1988 (Swedjemark, 1993).

Type of dwelling	Dwellings No.	Arithmetical mean, Bq/m^3	SD, Bq/m^3	Median, Bq/m^3	Maximum Bq/m^3
All dwellings	1360	108	179	53	3904
Detached houses	714	141	217	76	2765
Apartments	646	75	123	36	3904

No large national survey has been performed in workplaces. However, the radon concentration has been measured in a large number of workplaces, especially in schools and day-care homes. The results indicate that the average radon levels in workplaces do not differ much from those in homes. Thus, it may be assumed that the radon exposure to the population as such is equal independent of if the persons are at home or at work.

In Publication 65 from 1993 ICRP gives a dose coefficient of 0.021 mSv per year per Bq/m^3 as an average exposure during the year (equilibrium factor 0.4) (ICRP 65, 1993). This coefficient is based on epidemiological studies of lung cancer among miners. Applied to the average Swedish radon exposure 108 Bq/m^3 the average dose to the Swedish population would be 2 mSv/year (80% indoor occupancy). For people exposed to 1000 Bq/m^3 the dose will be 20 mSv/year and for those who were exposed to 80,000 Bq/m^3 the dose was more than 1 Sv/year.

However, as ICRP's dose coefficient is based on miners data these include results from smokers and non-smokers and the assumption that the exposure to radon is addita-

[1] The Swedish compulsory limits for radon gas activity concentration are for dwellings, schools, day-care homes etc. 200 Bq/m^3, for workplaces other than mines 400 Bq/m^3, all refer to the yearly average. The compulsory limit for mines is 2.5 MBqh/m^3 per year (equal to an exposure of 1500 Bq/m^3 as average during 1600 working-hours).

tive. The Swedish non-smoker study on radon and lung cancer and a recently published report on collaborative analysis of individual data from 13 European case-control studies of residential radon and lung cancer (Darby et al., 2005) shows that there exists a multiplicative synergetic effect of exposure to radon and smoking. That study clearly shows that the cumulative absolute risk of radon related lung cancer by age 75 years from same radon exposure is up to 25 higher for a smoker compared to a lifelong non-smoker. It means that the risk-equivalent dose from the same exposure varies with up to a factor 25 depending on smoking habits. Applied to the average Swedish radon exposure 108 Bq/m^3 the average dose to a never smoker is calculated to about 0.3 mSv/year, while the dose to a smoker is about 6 mSv/year.

In a survey, carried out by SSI, measurements of thoron (^{220}Rn) and thoron progeny were performed in 45 detached houses thought to be representative for Sweden. Thoron and thoron daughter concentrations in these dwellings were consistently low. The thoron progeny yearly average was 0.5 Bq/m^3 and the range 0.02-16 Bq/m^3 (Mjönes et al., 1996). The estimated average dose to the Swedish population is less than 0.1 mSv/year.

Radioactive elements in water

Naturally occurring radioactive elements in water can cause rather high doses through ingestion of food and drinking water. Radon escapes when the water is tapped and the gas mixes with the air. Used indoors as a rule of thumb if the radon concentration is 1000 Bq/L contribution of radon from the water results in radon levels of about 100 Bq/m^3. Thus radon from water when inhaled may be a cause of lung cancer.

Surveys of radon, radium and uranium in surface- and groundwater have been performed by SGU and SSI (Table 2).

Table 2. Naturally occurring radioactive elements in Swedish waters

Type of water	Radon-222, Bq/L	Uranium, µg/L [1)]	Radium-226, Bq/L
Lake and stream water (surface water)	< 1	0.001 – 0.8	0002 – 00.7
Sea water	< 1	1 – 3,3	0.002 – 0.003
Dug wells (groundwater from the soil aquifer):			
Arithmetic mean	43 [2)]	-	
Median	19 [2)]	-	
Normal in Sweden	10 – 300 max 3500	-	0.0002 – 0.09
Granite areas in Bohuslän	40 - 400	-	
Drilled wells (ground water from the bedrock aquifer):			
Arithmetic mean	203 [2)]	14.3 [3)]	0.09 [3)]
Median	86 [2)]	4.0 [3)]	0.01 [3))]
Sedimentary rocks, Närke	10 - 50		
Normal Precambrian igneous rocks	70 - 500		0.001 –0.25
Uranium-rich granites	300 - 4000, max 89000	max 268[3)]	0.05 – 0.8, max 7.5
Uranium-rich pegmatites	max 15000 - 30000		max 0.35 –2.5
Ground water in uranium ores:			
Lilljuthatten, Stenfjällen	2000 - 100000		max 6
Pleutajokk, Arjeplog	18000 - 55000		0.1 – 0.17
Population weighted mean	38[(4)]	1.8[(5)]	0.011[(4)]

[1)] The concentration of uranium is given, as µg U/L. For conversion to mBq/L see (Falk et al. 2004).
[2)] Results from 577 randomly chosen dug wells and 1276 drilled bedrock wells from all Sweden.
[3)] Results from 261 drilled bedrock wells sampled by SGU 2003-2004 (SGU, 2004).
[4)] Results from 375 public waterworks and 499 private drilled wells. (SSI-rapport 88-11, 1988)
[5)] Estimated from data from a study of naturally occurring radionuclides in Finland. (IRPA/NSFS. 1999).

The radon-222 activity concentration in surface water is with few exceptions very low, < 1 Bq/L. This is also the case for uranium and radium.. The average Swedish radon concentration in groundwater from the soil aquifer is 43 Bq/L (mean) and from the bedrock

aquifer 203 Bq/L[2]. Few analyses of the uranium concentrations in dug wells have been done. However, in 261 drilled wells sampled by SGU the average concentration was14 µg U/L[3]. As shown in Table 2 the range of concentrations in water of radioactive elements in bedrock water is very wide. The maximum concentrations so far encountered in Sweden is for radon 87 000 Bq/L, for uranium 268 µg/L and for radium-226 7.5 Bq/L. Sweden has approximately 100,000 dug wells and 300,000 drilled wells used by permanent residents.

In a survey, 2003-2004, the uranium and other radioactive material concentrations in drinking water from municipal water works that deliver water from soil or bedrock aquifers was conducted. Water samples from 256 municipalities were analyzed. (Falk et al. 2004). Nine water works show a uranium concentration above 15 µg/L. An estimated dose exceeding 0.1 mSv/year was found in two samples.

Table 3. Estimated internal doses from intake of natural radioactive nuclides.

Internal dose from food		**Internal dose from drinking water**		
Nuclide	**Mean, mSv/y**	**Nuclide**	**Mean, mSv/y**	**Max, mSv/y**
Uranium	0.0002	Uranium	0.0021	0.3
^{226}Ra	0.006	^{226}Ra	0.0022	1.6
^{210}Pb	0.022	^{210}Pb	0.0055	
^{210}Po	0.067	^{210}Po	0.0053	
^{228}Ra	0.011	^{228}Ra	0.0003	
		^{222}Rn	0.0024	5.7
		^{222}Rn$^{(*)}$	0.0704	165

(*) Dose from inhalation of radon released from water.

Internal exposure from naturally occurring radioactive elements.

The main contributor to internal radiation is potassium-40. Other natural radioactive isotopes that occur in the body are elements from the uranium and thorium series. Also cosmogenetic radionuclides as ^{14}C, ^{22}Na and ^{7}Be are found in the body.

The intake of these elements is via inhalation, food and drinking water. The internal dose from inhalation is small compared to the intake from food and water. Table 3 summarize the estimated dose from the nuclides that are of main concern. The internal dose for food is estimated from data in Europe presented in (UNSCEAR, 2000). The dose from ^{228}Ra is also estimated from (UNSCEAR, 2000), while the doses from ^{210}Pb and ^{210}Po are estimated from Finnish data (Mäkelainen I. Et al., 1999). Table 4 gives a summary of main internal doses from all naturally occurring radionuclides. Doses from ^{40}K and cosmogenetic produced radio nuclides are estimated from (UNSCEAR, 2000)

Dispersion of naturally occurring radioactive elements at burning of coal, oil and peat

Coal, oil and peat contain natural radioactive elements, which at burning are dispersed into the environment. The National Radiological Protection Board (NRPB) in Great Britain has studied the doses from released atmospheric particles from coal fired electrical plants (Smith, 2001.) The calculated dose to a typical member of the hypothetical critical group is in the order of 1.5 µSv/year and to an average individual 0.1µSv/year.

[2] According to the Swedish Drinking Water Act the limits for drinking water supplied to more than 50 persons or used for public or commercial purposes are for radon, 1000 Bq/l (unfit for use) and 100 Bq/l (restricted use). For uranium and radium is given a limit for the Total Indicative Dose (TID) which is 0.1 mSv/year. A person who daily has an intake of 100 µg U/l or 0,5 Bq ^{226}Ra/l receives a dos of 0.1 mSv/year.

[3] Uranium is chemically toxic. The WHO recommended limit for uranium in water at a daily intake is 15 µg/l.

Table 4. Summary of mean internals dose from naturally occurring radionuclides.

	Never-smoker	**Population**
Source	Mean, mSv/y	Mean, mSv/y
Food	0.11	0.11
Drinking water	0.09	0.09
Cosmogenetic	0.01	0.01
^{40}K	0.17	0.17
^{220}Rn	0.1	0.1
^{222}Rn (never smoker)	0.3	
^{222}Rn (population)		2.0
Sum	**0.8**	**2.5**

Total dose from natural radiation

The total dose to the Swedish popu-lation has been recalculated (Figure 1.) The average dose to the Swedish populations is 4.2 mSv/year. Due to the large difference in health effects of exposure to radon to never-smokers and smokers we in Figure 1 also show the total effective dose for never – smokers, which as calculated is 2.5 mSv/year**.**

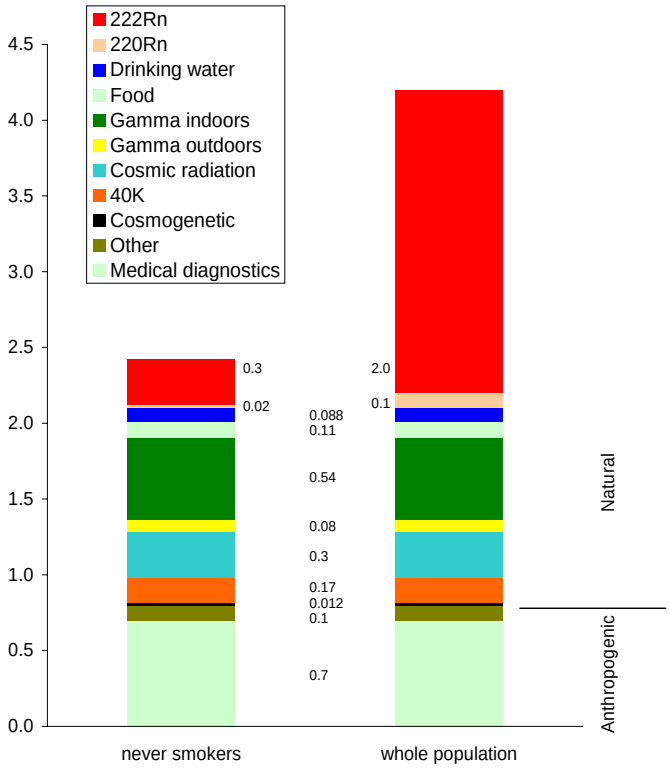

Figure 1. Total doses.

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Should radon be reduced in homes?

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The main results of a Norwegian cost-effect analysis is presented. This analysis is a follow-up study based on a model described in a paper published in 2003 (*Stigum et al 2003*). Stigum et al concluded that the cost per saved life by reducing the radon concentration in present homes to below 200 Bq/m^3 and by limiting exposure in future homes to below the same value, will be approximately $ 0.27 million (1.8 mill NOK) and the cost per life year saved was estimated to $ 20,000 (130,000 NOK). Based on a comparison with other health risks and the willingness from the society to pay for a life saved it was concluded that remedial and preventative measures against radon in homes were highly justifiable.

The total attributable fraction of lung cancers related to indoor radon exposure was estimated to be 11.2 %, and the preventable attributable fraction to 3.8 %. Based on the results of the European pooling analysis of epidemiological studies in homes (*Darby et al 2005*) and the results of the most recent large-scale radon surveys in Norway (*Strand et al 2001*), the attributable fraction is assumed to be somewhat higher – nearly 14 % (corresponding to an annual number of 280 cases at present) and the preventable fraction is assumed to be nearly 5 %. It was further assumed that all homes except blocks of flats above the ground floor level will have to be measured. However, the measurement costs can be reduced significantly by using a two-step measuring strategy as described by *Jensen et al (2004)* and in the *NRPA Radiation Protection Series no. 17 (1999).* Based on recent results of the cost and reduction effect of remedial measures it seems as though the cost of remedial measures have been underestimated in earlier analyses. Concerning preventative measures in new buildings, it is generally recommended by Norwegian authorities that such measures (radon membrane in combination with a passive system for ventilation of the ground and to outline pressure differences by installing a perforated channel system in the crushed rock under the concrete floor and connected to a pipe above the roof) are implemented in all new dwellings. By this strategy the costs of preventative measures will increase significantly, but the radon levels will also be reduced significantly in homes where the level would have been below 200 Bq/m^3 without any preventative measures. This will reduce the mean radon concentration in the future housing stock to significantly below the present annual mean level of 89 Bq/m^3 (*Strand et al 2001*)

Preliminary results of the follow-up analysis show that the cost per saved life is between $0.25 and $0.4 mill. which is less than the Norwegian society in other areas has been willing to pay in order to save a life.

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Dose and Dose Rates on Board Aircraft

A review of a recent EC-publication

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Abstract: A compilation of measured and calculated dose and dose rate results onboard aircraft has recently been reported by a EURADOS working group[1] (EC, 2004). More than 10000 experimental dose rate results and more than 300 experimental route dose values have been compared with calculated values for the same flight conditions. The results cover the period from 1990 to 2003 and cover then one solar cycle. In general the agreement between calculations and measurements is within 30 %. The report concludes that all codes and measurement methods have sufficient accuracy for radiation protection purposes when used as described.

Introduction

The annual average radiation dose to aircraft crew may become similar to or even larger than that of other occupationally exposed groups. The radiation field causing the dose is very complicated and involves a mixture of radiation types with energies not experienced in occupational exposure studies at ground level. Furthermore dose levels vary with altitude, geomagnetic latitude and the phase of the solar cycle. All those factors have highlighted the need to carry out thorough investigations of the radiation field at aircraft altitudes. This need is further underlined by the inclusion of cosmic radiation as occupational exposure under revised European Union Council Directive 96/29/EURATOM.

In this context, to guide authorities as well as companies concerned with exposure to cosmic radiation, the Article 31 Group of Experts (EURATOM Treaty), proposed in February 2000 to the Directorate DG XI Environment that a EURADOS[2] working group (WG) is installed with the aim to validate the existing dose rate data in flight altitudes. The report contains a short review of legal requirements on aircrew dose assessments, an overview of the physics of cosmic radiation as well as a discussion of the uncertainties involved not only in measurements onboard aircraft but also uncertainties accepted in the radiation protection field. The report was published by the European Community in late 2004 (EC, 2004). To make the understanding easier the main chapters include more than one hundred illustrations of which a few are included in this presentation.

The document includes results from a dozen different scientific groups that have performed measurements with for instance active instruments like ionization chambers, tissue equivalent proportional counters (TEPC), Bonner spheres, extended remcounters, GM counters, Si-spectrometers or with passive detectors like TLDs. track etched detectors, recoil- and fission track- detectors and bubble detectors (SDD). The measurement methods are described in an Appendix by those responsible for the measurements. In another Appendix the creators of the codes describe the codes used for the calculations. The available codes were: CARI, EPCARD, FREE, Pcaire and the algorithm of Pelliccioni. The

[1] The 28 working group members and contributors are found in the report

[2] European Radiation Dosimetry Group, see www.eurados.org

codes EPCARD and CARI are based on quite different approaches and are to a large extent independent of each other and have then been of particular interest to compare.

Results

The agreement is illustrated in figures such as Figure 1 and 2. The first shows the ambient dose equivalent rate as a function of altitude for the equatorial region (upper part) and for the Polar Regions (lower part) for a low solar activity -solar deceleration between 470 MV to 610 MV and the period 1992 to 2003. The calculated dose rate corresponding to those solar activities are shown as lines. The various symbols indicate the experimental results, which typically are within ±30%. In Figure 3 results from measurements on board Scandinavian airlines (Finnair and Scandinavian Airlines System) flights are compared with calculations. For the listed long distance flights the route dose varied between 25 and 70 μSv. The ratio of calculated and measured ambient dose equivalent is given for the different codes and is here in the interval 1.0 to 1.3. Similar results from measurements on board Icelandair are also found in the report. When other routes and measurement techniques are compared for the whole solar cycle, the major conclusion is that the agreement is within ±30% (two standard uncertainties).

Discussions and conclusions

Results from more than a dozen different groups making measurements on board aircraft have been compared with calculated results for the same conditions. The agreement is typically within ±30 %. A similar agreement is found for results calculated with the different codes. The working group therefore conclude that it should be sufficient to base aircrew dose estimates on calculations with the codes investigated.

For aircraft flying above 15 km continuous measurements of the exposure is requested (JAA, 2001). It has been argued from the aircraft industry that such instruments are no longer available. The report includes results measured with a number of different instruments. The conclusion from the WG is that all methods give sufficiently accurate results for applications in the radiation protection field. Although not specifically mentioned by the WG some of the active instruments are rather small and rugged and could most likely be useful for installations onboard aircraft. Even if the topic is quite complex, the large number of figures makes the reading easy and the report should be a valuable source of information for interested in this field.

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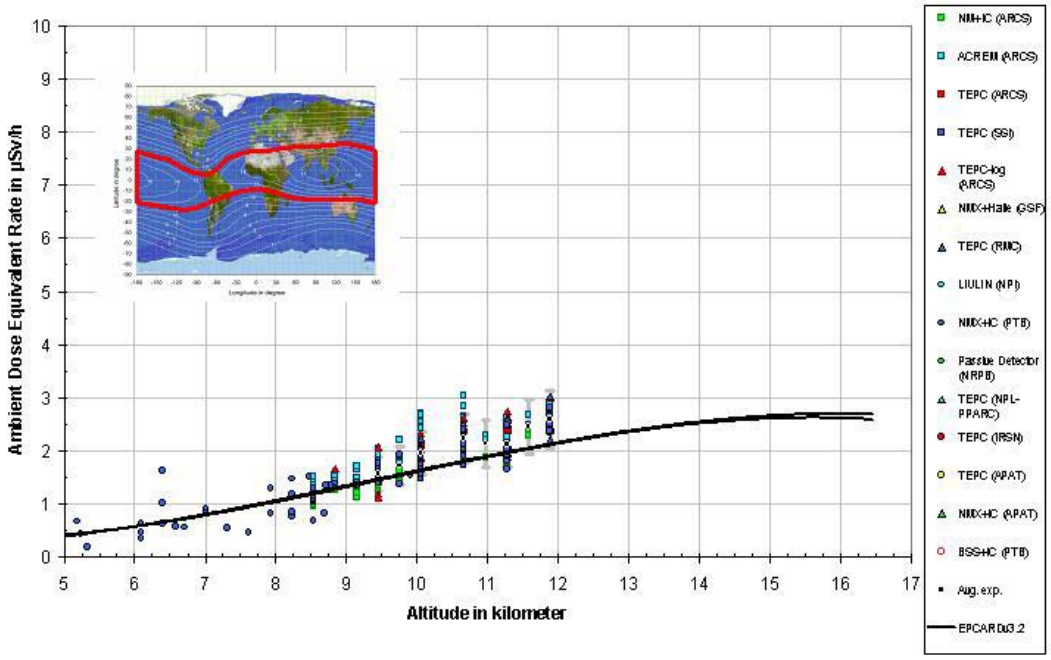
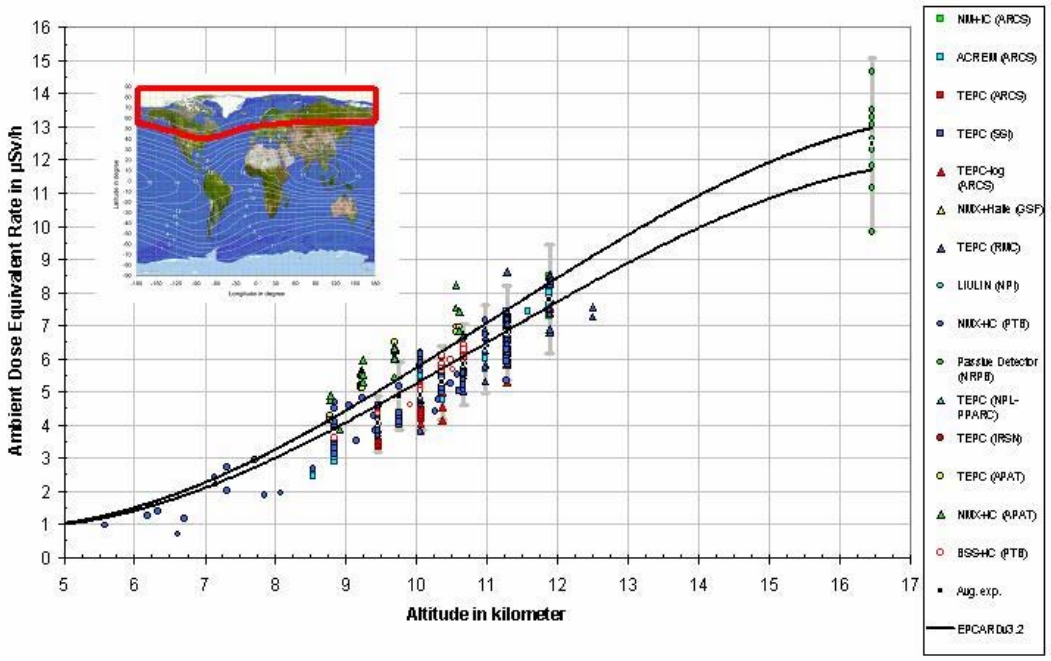


Ambient dose equivalent rate dH*(10)/dt vs. standard barometric altitude between May 1992 and May 2003 for vertical cut off rigidity rc ≥12 GV and solar deceleration potential in the range of 470 MV - 610 MV.



Ambient dose equivalent rate dH*(10)/dt vs. standard barometric altitude between May 1992 and May 2003 for vertical cut off rigidity rc ≤2 GV and solar deceleration potential in the range of 470 MV - 610 MV.

Figure 1. The ambient dose equivalent rate is shown in the Equatorial Region (upper part) and the Polar Regions (lower part) for the period 2002 and 2003 for different flight altitudes (EC, 2004). The insert shows the region for which the dose rate is applicable. At low solar activity the dose rate is high.

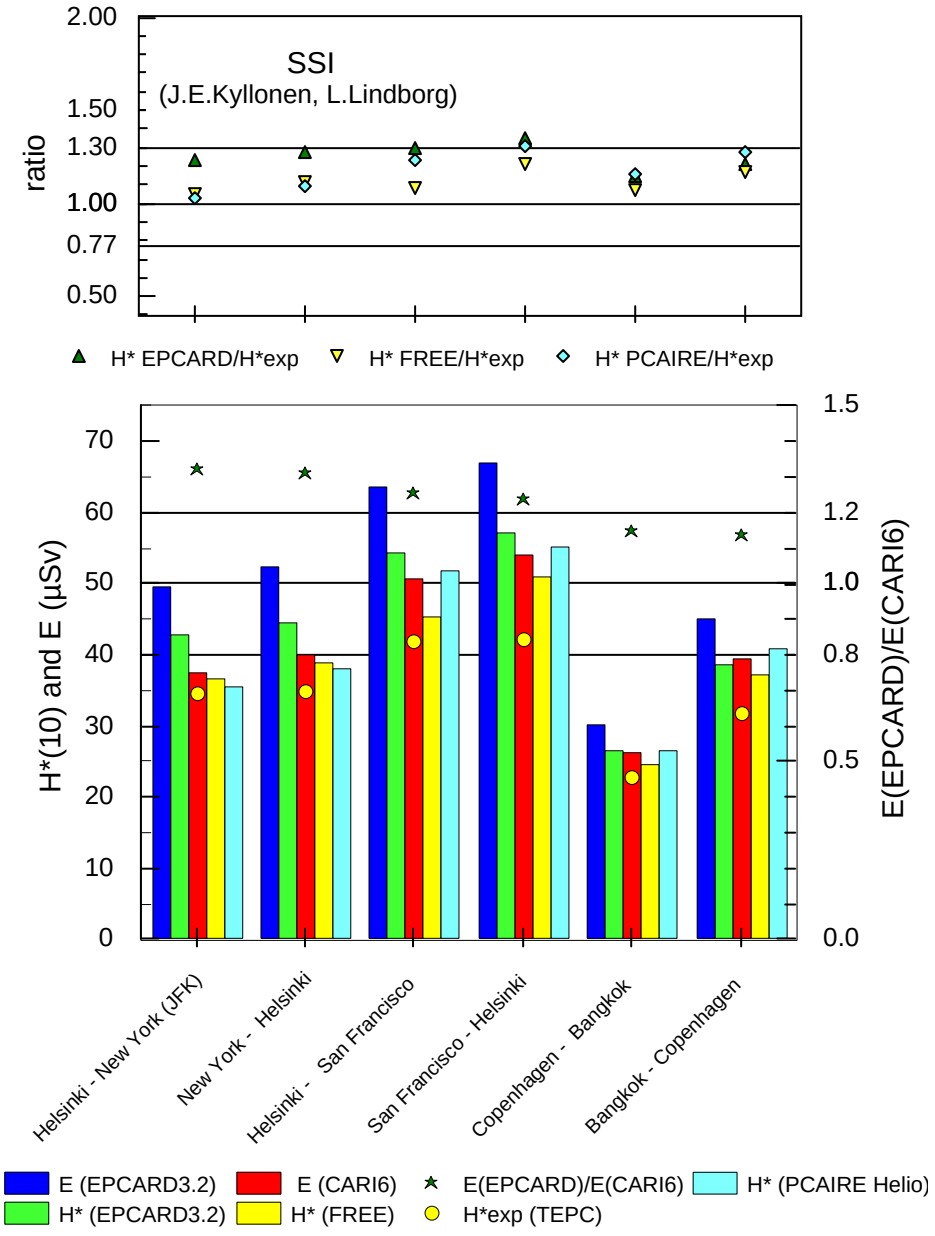


Figure 2. Calculated doses for flights of SSI (1998) and experimental data obtained with TEPC. The calculations were conducted with CARI, EPCARD, FREE and PCAIRE. The ratio of effective doses obtained with EPCARD and CARI6, respectively, is depicted together with a ± 20% band (right scale). Top of figure: Ratio of calculated operational quantity, *H**(10), and experimental data. An uncertainty band of ±30% (i.e. factor 1.3) is included (EC, 2004).

Radon in Finland: Building regulations and raising public awareness

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Abstract: New regulations of the National Finnish Building Code require consideration of radon risks and as a main rule radon technical design in the building permission documents. Slab-on-grade is the prevalent substructure in Finnish low-rise residential buildings. Without prevention the normal practices would result in high indoor radon concentrations in Finland. Guidance requires installation of protective sheet in the slab-on-ground foundation and a preparatory radon piping. Municipalities and STUK have launched a new campaign "Radon bee" (Radontalkoot) in order to increase the measurement and mitigation activity. In 2003-2004 40 municipalities have started the campaign. The campaigns have already resulted in 6000 new radon measurements.

Introduction

High radon concentrations in indoor air in low-rise residential buildings form an important national health problem in Finland. Without preventive measures indoor radon concentrations are clearly higher in new buildings than in the housing on average. The main reasons to this are the prevailing type of foundation is slab-on-ground (Fig 1). Use of very coarse, air-permeable and thick sub-slab filling sand layers and use of light-weight concrete blocks contribute also to the increase.

Table 1. Indoor radon concentrations in Finland

House type	Mean	> 200 Bq/m^3		> 400 Bq/m^3		> 800 Bq/m^3	
	Bq/m^3	%	N	%	N	%	N
Low rise	140	17.9	209000	5.0	59000	1.4	16000
Flats	70	1.6	16000	0.8	7000	0.3	3000
All	120	12.3	225000	3.6	66000	21.0	19000

In many areas tens of percent of low rise residential buildings exceed the reference level for new buildings, 200 Bq/m^3. On average 18 % of low rise buildings exceed the level (Table 1). Taking into account that the probability to high concentrations, without radon prevention, is high and the facts that new practices promote in inflow of radon-bearing soil air into living spaces, the only working and practical advice is that in all building the risks of radon should be taken into account.

A study project in 1999-2002, funded by the Finnish Technology Agency "Radon-safe foundation, moisture prevention and air exchange in a healthy building" aimed in finding simple construction practices, which can be utilised for prevention of the leakage flows in slab-on-ground foundation. In this project a new construction for a radon-tight joint between the foundation wall and floor slab was developed. The work lead up to the renovation of guide written on radon-safe building, in co-operation with the Ministry of Environment. In 2003-2004 also the regulations of the national Building Code were renewed. This paper presents the key features of both the practical guide and the building regulations.

The second part of this study presents the key features of a new campaign “Radon bee” (Radontalkoot) which aims at increasing the indoor radon measurement and mitigation activity of the citizens.

New legislation

The National Building Code of Finland includes both regulations which are mandatory and guidelines which provide acceptable solutions in conformity with regulations.

In 2003 a new Building Code, part D2, Indoor climate and ventilation in buildings was issued. The Code gives a design value for indoor radon concentration in a form of mandatory regulation.

> **Quality of indoor air. Regulation:** New houses must be designed and constructed so that indoor air is free of harmful concentrations of gases, particles or microbes or odours which decrease the indoor comfort. Design value for radon is 200 Bq/m^3.

In 2004 a new Building Code, part B3, Substructures, was issued. The main regulation concerning radon is as follows.

> **Radon. Regulation:** In the design and construction work, radon risks at the construction site shall be taken into account.

The following guidelines give several detailed instructions concerning building practices and building permissions.

> The limit value 200 Bq/m^3 is generally exceeded in the most part of Finland, if no countermeasures are taken.
>
> A radon-technical design may be left out only in the case the local radon surveys clearly show that the radon concentration in residential buildings is consistently below the permitted maximum value.
>
> If radon is not taken into account in design, written grounds for that shall be the attached to the design documents of the building project.
>
> The original soil and soil brought elsewhere to the site for land filling, as well as drainage gravel, always have an impact on the risk of radon in the building ground. A thick layer of gravel filling alone can produce radon concentration exceeding the limit value.
>
> The radon concentration inside a building can be significantly influenced by the selection of base floor structures and foundation method.

Later guidelines of B3 give practical information on radon safe foundations.

By ensuring tightness of base floor and walls resting against ground, the passage of air containing radon into the building can be prevented. Crawl space and tight uniform slab are radon safe constructions. If the foundation wall and slab are constructed separately - this is the prevailing method in Finland - the following countermeasures should be carried out:

- seal the joint between floor slab and foundation wall
- install radon piping, the piping will be activated if sealing work does not result in a low radon concentration
- seal lead-throughs

Guidance for radon-safe building

New guidance for radon safe building was issued in 2003 as RT Building Information File, RT-81 1079, Radon prevention. The guide was published by Building Information Ltd and was renewed in co-operation with the publisher, Ministry of Environment, local building authorities, companies and STUK. The guide focuses especially on sealing of the joint between floor slab and foundation wall using bitumen felt.

Implications of new regulations and guidance

- Building authorities are today more aware of radon and have made improvements in practices concerning radon documents in building permission process.
- Building companies have gained experience in the need for radon safe building and have renewed their radon practices.
- New house-owners are more aware of radon, test their new dwellings and require the companies e.g. for activating the radon piping if radon concentration exceeds 200 Bq/m^3.
- Companies provide radon products as piping packages for new construction and bitumen felt for sealing work

Radon bee

Radon bee is a new campaign directed to municipal health authorities aiming at increased radon testing activity and radon awareness and knowledge and finally at increased radon mitigation activity. Radon bee is co-ordinated by STUK. Ministry of Social Affairs and Health promotes and funds the projects.

Local authorities have the key role in the campaign. They receive the orders and deliver the dosemeters to the house-owners. Radon dosemeters have been offered to house-owners by half price.

Figure 1 Logo of radon bee. Upper text: Clean your house of radon. Lower text: Radon bee, Test, fix and check !

The role of STUK in the campaign includes:

- Planning the campaign in co-operation with the local authorities
- Co-ordination of the campaign advertisements in local journals
- National communication
- Lectures in citizen evening
- Remediation training of local building companies
- Follow up and reporting
- www-pages on radon campaigns at www.stuk.fi

Radon bee was started in a pioneer campaign. In autumn 2004 18 new campaign were started . The campaigns are today running in 40 municipalities and altogether 6000 new radon measurements have ordered. The number of measurements ordered ranges from 60 to 1000. In some municipalities nearly as many new measurements have been carried out as carried out altogether in previous 15 years. The campaign aims at training of local building companies in radon mitigation and provides a free one day training day. Altogether 110 participants have attended the four training days in 2004 and 2005.

Radon bee will be continued in 2005-2010 aiming at reaching most of the municipalities.

Radon mitigation in large buildings in Finland

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Abstract: This is a case study of radon mitigation in seven large buildings, in school and industrial buildings. Objects are situated in high radon risk areas. Highest indoor radon concentration at the beginning was 7000 Bq/m^3, and after mitigation below action level 400 Bq/m^3 during working hours. Several techniques were used, as well as combinations of these.

Introduction

Objects are large buildings, schools, warehouses, office buildings and industrial buildings. They are situated in high radon risk areas in Finland. Floor space is mainly from thousand m^2 to tens of thousands m^2. Radon measurements have been performed in the ground floor, and in some cases also in the first floor. Only in one case, the working place was in the basement. First, the two-month measurements were done with passive radon detectors, and after that, mainly one-week measurements with Alpha Guard or Pylon AB-5 measuring equipments, and then the measuring period was one hour.

Results and discussion

The first case is a warehouse and office building having several storeys. It is been built on granite rock and there are crushed aggregate under the floor. The floor consists of several concrete slabs. Supporting pillars go through the slab. There is also automatic intake/exhaust ventilation. The radon concentration was higher than the action level, 400 Bq/m^3, in two places, in the packing and in the planning department. In the beginning, the radon concentration in the packing department was 580 Bq/m^3. Timing of ventilation was adjusted to start three hours before working hours. The radon concentration was mitigated below the action level. In the planning department, in the middle of ground floor, the radon concentration was still 820 Bq/m^3, ranging 370-1370 Bq/m^3. They sealed up chinks around pillars, and installed seven pipes through slab for sub-slab suction alongside the inside walls. The outflow was adjusted to 0.5 m^3/h/floor m^2. The final result for the radon concentration was 50 Bq/m^3, ranging 10-160 Bq/m^3, and radon concentration during working hours was 20 Bq/m^3. If the fan stops, an automatic mechanism sends an alarm to service man. The total costs were 20 000 €.

The second case is an industrial building. It contains two factory sheds and an office sector with two storeys. It is built on esker. The walls are made from blocks of lightweight concrete. The floor consists of several concrete slabs. The supporting pillars go through the slab. There are wide chinks between the wall and the floor, and around the pillars. In the beginning, the radon concentration was between 600-4500 Bq/m^3, and the highest value was measured at an office. They sealed up chinks between the wall and the floor, and also around the pillars. The timing of ventilation was adjusted to start three hours

before working hours. The radon concentration was mitigated below the action level in one factory shed. In the other, the ventilation was adjusted to overpressure. There is now a risk of moisture in constructions of the building. Outside the office sector, they made a radon well. After the mitigation, the radon concentration was between 30-200 Bq/m^3 during working hours.

The third case is an old office building. It is been built on esker. There is gravitational ventilation. There is an archive in the basement. The building has concrete floors. The floors in the basement are situated on several levels. There are also old pipe channels. In the beginning, the radon concentration in the archive was 7200 Bq/m^3, ranging 4660-11500 Bq/m^3. The ventilation was tried to get more effective in the basement. The fan was situated in the archive and blew air into an old brick flue. The flue leaked into the ground floor, and the radon concentration increased to 2830 Bq/m^3 in the office. Now a sub-slab suction has been installed there, and the fans are on the roof. There is also a liner pipe installed inside the flue. The radon concentration in the archive is now 950 Bq/m^3, allowing 600 hours of working time during a year. In the office radon concentration is 390 Bq/m^3, and during working hours 290 Bq/m^3.

The fourth case is a school. It has been built on esker. The basement is partly on subsurface of ground level from all sides. There is a concrete floor. In the beginning, the radon concentration in the kitchen and canteen was 1400 Bq/m^3, ranging 750-5200 Bq/m^3. During renovation they set up a new automatic ventilation. The sub-slab suction from three points was built so that pipes go under the slab to an old heating plant, and the fan blows into an old brick flue. The flue is tight. The brick flue is cold and the upper part of it will weather in time because of moisture. After the mitigation, the radon concentration is 200 Bq/m^3, ranging 10-640 Bq/m^3 and the radon concentration during working hours is less than 100 Bq/m^3.

The fifth case is also a school. It has been built on esker. There is an old building with several storeys in connection with a new building with two storeys. The new one is a long building built along an esker. The whole back wall of the ground floor is against the hillside. In the beginning, the radon concentration was varying between 500-7000 Bq/m^3. The radon concentration increases when going away from the old building. During renovation they have installed a new automatic ventilation. The sub-slab suction is built so that a long drainpipe goes along with the base. This construction method is not working equally in all sections of the base. A better way would be a tight pipe line along the building with drainpipes going crosswise, and five 10-mm holes going into the tight pipe at the point of contact. The ventilation starts up on work days three hours before working hours, except after weekend it starts up on Sunday evening. In the classroom with the worst radon concentration, the average value of radon concentration is 3000 Bq/m^3, ranging 100-7000 Bq/m^3, and during working hours 260 Bq/m^3.

The sixth case is a school on esker with two storeys. The whole back wall of the ground floor is against the hillside. In front of the ground floor there is a narrow courtyard, and the hill slopes down further. In the beginning, the radon concentration was varying between 300-5000 Bq/m^3, in the ground floor. The radon mitigation was done in several steps. The methods used were sealing, adjustment of ventilation and radon wells. They built four radon wells alongside the walls, one at the both gable, and two underneath the building. The wattage of two the fans are 75 W, and the other two 120 W. The outflow of the radon wells varies between 16-96 m^3/h. The radon concentration during working hours varies between 70-230 Bq/m^3 in the classrooms, but in the store, in the rear end of

the ground floor, there is still a radon concentration of 1990 Bq/m^3. The best place for the radon well would be in the upper part of the hill.

The seventh case is a vocational school. There are six buildings on top of a narrow, sharply outlined esker. There is a regional heating plant, and the heating pipe conduit connects the buildings together. In the beginning, the radon concentration was varying between 30-16 000 Bq/m^3. The radon mitigation was done in several steps. The methods used were sealing, adjustment of ventilation, sub-slab suction, and radon wells. They have built six radon dwells between the buildings. Because radon wells did not have effect to the whole area, they have also built a sub-slab suction in three of the houses. They have suction also from the heating pipe conduit. The radon concentration during working hours is now less than 400 Bq/m^3 in five buildings. In one building, the radon concentration is higher than the action level. Radon comes in very easily through structures of building. The radon concentration is varying between 10-2450 Bq/m^3, and it depends on wind direction. This building is out of use, because they have enough room in other buildings.

In Finland they have used mainly same radon mitigation methods in large buildings, as in small buildings, as well as combinations of these. Radon mitigation has succeeded very well. It was difficult to find out real costs. In many cases the radon mitigation was one part of wide renovation, and did not have own price. Employers own staff did work in some cases, and even then they did not count the cost of radon mitigation.

Radon in Norwegian dwellings

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In this presentation, the main results of the latest large-scale surveys of radon in Norwegian dwellings are presented. In the heating season 2000/2001 radon measurements were carried out in 29,000 dwellings in 114 out of 435 Norwegian municipalities. The dwellings were randomly selected, and the sample size in each municipality varied between 10 % of the housing stock in rural areas to 2 % in the most densely populated municipalities. The measurements were made by etched track detectors (Cr-39), one in each dwelling. The detectors were located in the main bedroom or the main living room. The integration time was two months, and the measurements were restricted to the heating season (October –April).

Based on the results of the survey, the population weighted annual mean radon concentration was calculated to 89 Bq/m^3. The radon level in 9 % and 3 % of the dwellings exceeded 200 and 400 Bq/m^3, respectively. In these calculations it has been taken into account that the radon concentration is higher in the winter than in the summer. The corrections are based on results of an earlier survey where measurements were evenly distributed throughout the whole year. Generally, indoor radon concentrations in Norway are twice as high in the winter season (October – April) than in the summer season.

Comparisons of the results of the 2000/2001 survey and the results obtained from measurements performed in the same municipalities during the nationwide 1987-89 radon survey (Strand et al, 1992) revealed that the average radon concentration in the present housing stock is 70 % higher than twenty years ago. This increase can be explained by several factors, of which the following three are considered to be of main importance: 1) a large proportion of old houses have been retrofitted, and after the energy saving campaign of the beginning of the seventies, modern houses have been made more energy efficient, often leading to a reduction of the air exchange rate, 2) extended use of highly permeable light expanded clay aggregate blocks in the foundation walls which started around 1970, and 3) in modern homes, a higher proportion of bedrooms and living rooms are located on the ground floor or the basement compared to older homes.

Up to now, it is assumed that 80,000 homes in Norway have been monitored, and based on survey data, it is estimated that approximately 175,000 homes are exceeding the recommended action level of 200 Bq/m^3. So far, only 8000 (4.6 %) of these dwellings have been identified by indoor measurements and less than 3000 have been mitigated. There are other papers describing the Norwegian radon mapping and remediation program (Jensen et al, 2004), and this short presentation will focus on the results of the latest large-scale surveys, only. A more detailed discussion of the results is in preparation and will be published later.

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Natural uranium in drinking water and bone

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Abstract: We studied 146 men and 142 women aged from 26 to 83 years who had used drinking water originating from wells drilled in bedrock for 13 years on average, in areas with naturally high uranium content in bedrock. Two biochemical indicators of bone formation and a marker for bone resorption (CTx) were analyzed. The primary measure of uranium exposure was uranium concentration in drinking water. The median uranium concentration in drinking water was 27 µg/l and the median total committed equivalent dose of bone surfaces from uranium isotopes in drinking water was 3.3 mSv. There was some suggestion that elevation of CTx could be associated with increased uranium exposure in men, but no similar relationship was found among women. Hence, bone may be a target of chemical toxicity of uranium in man.

Introduction

High natural uranium concentrations have been found in private drilled wells located mostly in the granite areas of southern part of Finland. Uranium accumulates in bone in continuous exposure, affects bone metabolism in laboratory animals, and when ingested in drinking water, increase urinary excretion of calcium and phosphate, important components in the bone structure. The kidney has been considered the main target organ of chemical toxicity of uranium in man but effects in other tissues or organs remain poorly known. The aim of this study was to assess whether long-term uranium intake through drinking water affects the bone turnover in human.

Methods

Study population, sample collection and preparation: The source population was identified from the drinking water database of STUK, including more than 5,000 drilled wells. The final study population consisted of 288 persons from 179 households (Table 1). The water, urine and non-fasting blood samples were collected between September and December, 1999. The study persons brought the water and urine samples collected overnight to the laboratory in the morning. At the same visit, blood samples were taken.

Table 1. Description of the study population.

	N		Mean		Median		Min		Max	
	Men	Women	Men	Women	Men	Women	Men	Women	Men	Women
Age (years)	146	142	53	52	54	53	26	28	78	83
Body mass index (kg/m^2)	143	128	26	25	25	24	20	18	35	41
Number of deliveries	.	82	.	2	.	2	.	0	.	6
Duration of the use of drilled well (years)	146	142	13	13	11	11	2	1	34	34

Uranium exposure assessment: Uranium in drinking water and urine were analyzed with inductively coupled plasma mass spectrometry. The primary measure of uranium exposure was uranium concentration in drinking water. In addition, daily intake of uranium from drinking water, cumulative intake from drinking water and uranium concentration in urine were measured.

Outcome variables: Serum osteocalcin and amino-terminal propeptide of type I procollagen (P1NP) were used as indicators of bone formation. Osteocalcin was analyzed using an immunoradiometric assay and P1NP with a commercial RIA. Serum type I collagen carboxy- terminal telopeptide (CTx) was used as an indicator of bone resorption. CTx was analyzed with an enzyme immunoassay.

Statistics: The robust regression method in Stata/SE 8.1 was used. Robust regression assigns a weight to each observation with lower weights given to possible influential observations. The analyses were performed separately for men and women:

$$\ln(y)=\alpha + b_1 \ln(x) + b_2\, age + b_3\, age^2 + b_4\, smo \qquad \text{(for men)}$$

$$\ln(y)=\alpha + b_5 \ln(x) + b_6\, agecat + b_7\, smo + b_8\, estro \qquad \text{(for women)}$$

in which y=indicator of bone metabolism, α=constant, b=regression coefficient, x=continuous uranium exposure, agecat=age category (45-55, 55-65, and >65 years), smo=current smoking status, estro=use of estrogens.

Results and discussion

Background: For men, the levels of the markers of bone turnover decreased significantly with age (all P-values were <0.01 for associations between linear age and all outcome variables). Osteocalcin in men tended to decrease with age until about age of 60 years, after which bone turnover appeared to increase gradually (P=0.06 for quadratic age variable). For women, the levels of osteocalcin, P1NP, and CTx were highest in the age group of 55-65 years but the differences between the age groups were not statistically significant. Estrogen use was associated with significantly decreased levels of osteocalcin, P1NP, and CTx. Background levels of CTx for men and women are shown in Figure 1.

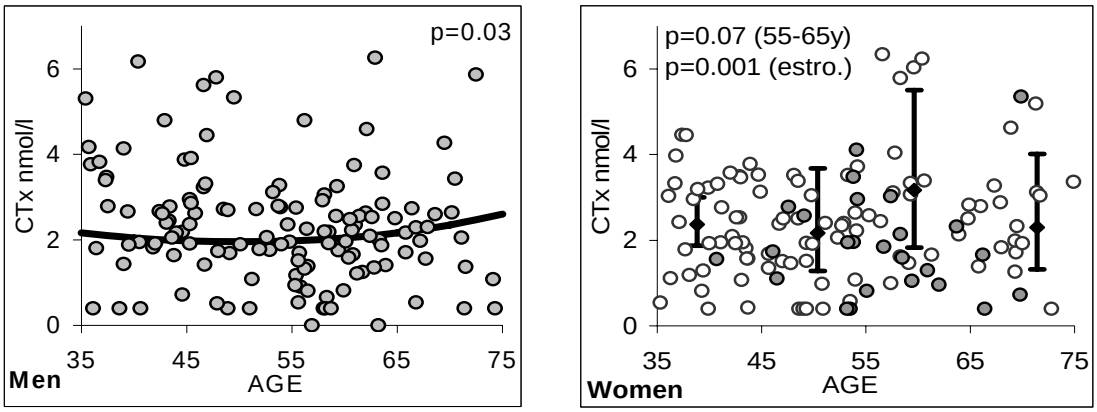

Figure 1. Background levels of the marker of bone resorption (CTx). For men the curvature line and the p-value represent the estimates of squared age variable. For women open circles represent women who do not use estrogen and filled circles women who use estrogen. Error bars are 95% confidence intervals of age groups. P-values are given for the age group of 55-65 years in comparison with age group of <45 years and for the estrogen users in comparison with non-users.

Uranium exposure: The uranium concentration in water varied from 0.001 to 1920 µg/l (Table 2), with 27% of the concentrations exceeding 100 µg/l and 59% above 15 µg/l. The median total committed equivalent radiation dose of bone surfaces from uranium isotopes in drinking water was 3.3 mSv (max 911 mSv), based on the cumulative uranium intake and the average uranium isotope activity ratios (^{234}U: ^{238}U =2) measured in Finnish drilled well waters and dose conversion factors (^{234}U: $7.4 \cdot 10^{-7}$ Sv/Bq; ^{238}U: $7.1 \cdot 10^{-7}$ Sv/Bq).

Table 2. Levels of the uranium exposure indicators. Doses are radiation doses received from uranium isotopes in drilled wells. N=288

			Percentile		
	Mean	Median	25th	75th	Max
Uranium in drinking water (µg/l)	119	27	6	117	1920
Daily intake of uranium from drinking water (µg)	214	36	7	207	4128
Cumulative intake of uranium from drinking water (g)	1.27	0.12	0.02	0.66	33.1
Uranium in urine (µg/l)	0.34	0.07	0.02	0.35	4.54
Uranium in urine (µg/mmol creatinine)	0.058	0.011	0.002	0.057	0.57
Committed equivalent dose of bone surface (mSv/yr)	2.12	0.36	0.07	2.05	40.9
Effective dose (mSv/yr)	0.13	0.02	0.005	0.13	2.60
Cumulative committed equivalent dose of bone surface (mSv)	35.1	3.3	0.6	18.1	911
Cumulative effective dose (mSv)	2.22	0.21	0.04	1.15	57.8

In men, uranium exposure was associated with elevated CTx levels (Figure 2). Some evidence for an association between increased levels of osteocalcin and uranium concentrations in drinking water was also found. Levels of P1NP were not associated with uranium exposure. Uranium concentrations in urine were not associated with the markers of bone turnover. In women, uranium exposure was not associated with any indicators of bone turnover (Figure 2).

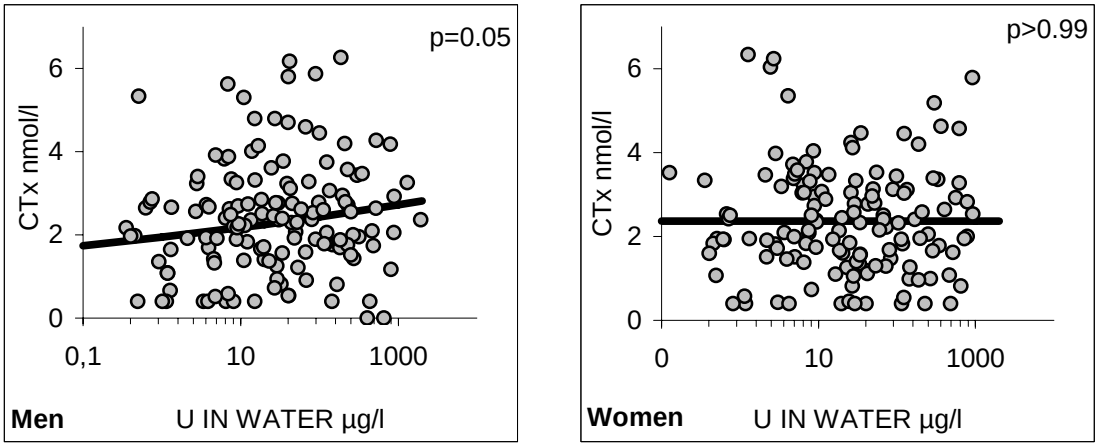

Figure 2. Associations between the marker of bone resorption (CTx) and uranium exposure. Regression lines and p-values were taken from the robust regression models.

Conclusions

In men, chronic uranium exposure indicated by uranium level in drinking water as well as daily and cumulative uranium intakes tended to be associated with the increased levels of bone resorption marker CTx and to a lesser degree of bone formation marker osteocalcin. This finding may indicate that bone is a possible target of chemical toxicity of natural uranium.

In contrast to men, no statistically significant associations with uranium exposure and the measured bone turnover markers were observed in women. In women, subtle effects may be masked by other strong determinants of bone turnover, such as menopause and hormone use which could not be effectively controlled in the present study.

Substantial variation in age complicates the interpretation of the results because several age dependent factors influencing the bone turnover may mask possible effects of uranium. As was seen in this study, the levels of bone turnover markers remain approximately stable from 25 years to menopausal age in women and to 65 years in men. Obviously focusing on limited ages would facilitate the interpretation of the results.

In summary, we found some evidence for an association between increased bone turnover and exposure to natural uranium through drinking water among men. The fact that similar effects were not observed in women may be due to other stronger factors in bone metabolism of women may mask the effects of uranium. This study suggests that in addition to kidneys, bone may be another target for uranium toxicity.

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^{210}Po and ^{210}Pb in seals from the Baltic Sea and Lake Saimaa, Finland

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Abstract: ^{210}Po and ^{210}Pb are members of the ^{238}U decay chain. ^{210}Po is an α-emitter with a half-life of 138 days, while its grandmother, ^{210}Pb is a β-emitter with 22.3 year half-life. In the atmosphere ^{222}Rn formats its decay products ^{210}Po and ^{210}Pb. These nuclides are deposited on to the surface of land and sea and thus enter the food chain. The naturally occurring radionuclides ^{210}Po and ^{210}Pb are important because their great contribution to radiation dose to human and other species.

As top predators in the aquatic food chain, fish-eating seals are vulnerable to the accumulation of contaminants. In the Regional Laboratory in Northern Finland, measurements of ^{210}Po and ^{210}Pb activity concentrations in seals from the Baltic Sea and in ringed seals from Lake Saimaa have been performed. Concentrations of ^{210}Po and ^{210}Pb in seals were determined in muscle, liver, kidney and spleen. The results of ^{210}Po and ^{210}Pb activity concentrations and the ratio of ^{210}Po / ^{210}Pb in these samples are presented.

The effect of uranium in drinking water on the uranium concentration in urine and hair

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Abstract: In Finland exceptionally high concentrations of natural uranium in drinking water have been measured in the granite areas in southern Finland. Consequently high concentrations of natural uranium have been observed in the urine and hair samples of people using water from their own drilled wells. The uranium contents in drinking water, overnight urine and rinsed and digested hair samples were determined in some families which have used water rich in uranium from their own drilled wells. The determinations were carried out by ICP-MS. Different methods were used to decrease the uranium concentration in drinking water. The highest uranium concentration in drinking water was 11000 μg/l, in urine 47 μg/l and in hair samples 820 μg/g.

INTRODUCTION

In Finland exceptionally high concentrations of natural uranium in drinking water have been measured in the granite areas in Southern Finland. The average uranium concentration in drilled well water has been established as 21 μg/l (max. 800 μg/l) in a random sampling study recently carried out in Finland. Much higher uranium concentrations (max. 12 400 μg /l) have also been measured but rarely (Vesterbacka et al. 2001, Salonen 1994). Consequently, high concentrations of natural uranium have also been observed in the urine and hair samples of people who consume water from their own drilled wells. The uranium content in these excretion pathways was noticed to correlate with the uranium intake, particularly at elevated levels where drinking water was the major source of exposure to uranium (Karpas et al. 2005). In our present study, the effect of the drinking water on the concentration of uranium in urine and hair was studied. In this paper the preliminary results are presented.

METHODS

The uranium concentration in drinking water, overnight urine and hair samples were determined among a few families that had have used uranium rich water from their own drilled wells.

The radiochemical analysis of uranium in drinking water was carried out by concentrating uranium from water using iron hydroxide scavenging. The precipitate was dissolved in concentrated hydrochloric acid and the uranium isotopes were separated from other radionuclides using an ion exchange resin (Dowex 1x4, 50/100 mesh). Uranium was finally co-precipitated with CeF_3 and counted with the AlphaAnalyst alpha spectrometer (Canberra, US). ^{232}U was used as a chemical yield tracer.

Overnight urine samples were collected in clean plastic bottles and then transported into the laboratory where subsamples were removed into an acid-cleaned plastic tubes for

uranium analysis. Uranium was analysed by high resolution ICP-MS at Consulting Engineer Paavo Ristola Laboratory in Finland and by ICP-QMS equipped with a flow injection system at the Geological Survey of Israel (Lorber et al. 1996).

Hair was collected by cutting about one cm^2 area near the ear and close to the scalp. Approximately 1 g of hair sample was weighed directly into centrifuge tubes. Sample was washed according to the IAEA standard procedure using acetone and distilled water (Chatt et al.(1985). A dried sample was weighed into a crucible to be dry ashed. The ashing temperature was raised slowly during 20 hours to the final temperature of 500 ºC after which the ashing was still continued for 3 hours. The ash was transferred to a beaker with 6 ml of a 2:1 mixture of HNO_3 (65 % Baker pro analysis grade):H_2O_2 (30 %). The beaker was covered with a watch-glass and heated to the dryness on a hot plate. The sample was then removed from the hot-plate and cooled slightly and 2 ml of H_2O_2 were added. The sample was again heated to dryness and allowed to cool before adding 2 ml of concentrated HNO_3. The sample was then heated to dryness. That procedure was repeated until the precipitate was white. The remaining residue was dissolved in 4 ml of HNO_3 (65% Merck Suprapure) and transferred to acid-cleaned 10-ml plastic tubes for uranium analysis by ICP-MS. The hair samples were measured at Consulting Engineer Paavo Ristola Laboratory and some of samples at the Technical Research Centre in Finland (VTT).

RESULTS AND DISCUSSION

The uranium concentration in drinking water, urine and hair are presented in Table 1. The highest uranium concentration in drinking water of the families studied here was 11000 µg/l, in urine 45.5 µg/l and in hair samples 820 µg/g.

The variation of uranium content in a drilled well water of the subject family 2 is presented in Table 2. It indicates high temporal variation of uranium in thie well water. This has affected considerably the uranium concentration in the urine and hair samples of two family members, -the wife and the husband. They had used this source of drinking water for six years. The well water also contained high amounts of radon and other radionuclides from the uranium series.

Alternate means were used to decrease the uranium concentration in drinking water. In the first test bottled mineral water was used for two weeks instead of the tap water from their drilled well, and later on uranium was removed from drilled well water using an ion exchange method. STUK delivered the mineral water to be used as drinking water during the period from October 28th, 2003 to November 11th, 2003. After this test period the family again started to use their own drilled well water. Before starting to use mineral water the family members collected overnight urine samples. After starting the usage of mineral water the urine samples were collected after 4, 7 and 14 days and once more a week later when they had started to use again their own well water. The hair samples were taken on November 11th, 2003, when the bottled water had been used for two weeks and before changing it to drilled water again. An anion exchanger was installed for removing uranium from the well water on June 19th, 2004. After that the uranium content both in urine and hair was measured. The follow-up measurements will continue for several years. Thus valuable information on the retention and excretion of uranium will be obtained by the measurements performed before and after the installation of the ion exchanger.

The high uranium concentration in drinking water had a significant effect on the concentration of uranium in urine and hair. When drilled well water of the subject family 2 was changed to mineral water, a sharp decrease in the concentration of uranium in urine was observed four days after the change took place. The steep decrease continued over seven days. After that the decrease was much more gradual.

The husband of the subject family 2 probably did not drink much water from their own drilled well after two weeks test because of travelling related to his work. This can be seen in figure by samples collected after November 11th. The wife change back to drink more of their own drilled well water. This is evident by a sharp increase in the uranium content of her urine sample. The immediate effect is due to stopping the supply of ingested uranium and the slow decrease could be due the body stocks beginning to excrete.

Uranium concentration in the private wells drilled in bedrock may vary considerably over time, and therefore a spot sampling is not necessarily a good indicator of the long-term uranium exposure.

Table 1. Uranium concentration in drinking water, urine and hair of the three subject families.

Family	Person	Date	Water µg/l	Urine µg/l	Hair µg/g
Family 1	Husband	11.3.2002		0.83	
		5.3.2003			18
		10.3.2003	3310	0.18	
	Daughter	17.6.2002		5.5	
		11.3.2003	3310	1.51	
Family 2	Wife	16.2.2001	80	3.2	
		11.3.2002		5.5	
		17.12.2002			625
		15.3.2003		26.2	
		28.3.2003			821
		14.6.2003	2500	31.9	
		4.11.2003	*	9.9	
		6.11.2003	*		541
		19.5.2004	90	0.5	
		24.5.2004			235
		9.6.2004	2500	12	
	Husband	16.2.2001	80	6.3	
		9.3.2002		7	
		15.3.2003		21.3	
		28.3.2003			
		1.4.2003			610
		14.6.2003		6.3	
		16.6.2003	2500		
		4.11.2003	*	13.2	
		6.11.2003	*		530
		19.5.2004	90	2.4	
		9.6.2004	2500	4.4	
Family 3	Husband	13.3.2002		0.03	
		13.11.2002			8

* The members of the family used bottled water instead their own drilled well water.The uncertainty of the results are 10 - 15 %.

Table 2. Temporal variation of uranium concentration in the drilled well water of the subject family 2.

Date	Uranium µg/l
16.2.2001	80
16.6.2003	2500
27.10.2003	9200
4.11.2003	9400
11.11.2003	9500
17.11.2003	11000
19.5.2004	90
8.6.2004	2500
9.6.2004	2500

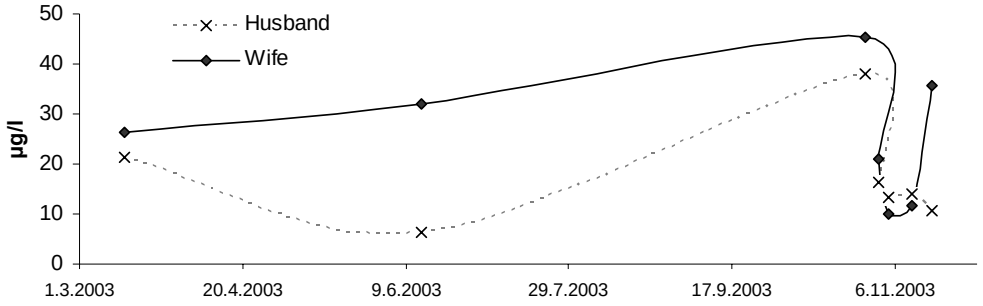

Fig1. The concentrations of uranium in urine samples of the members of the subject family 2.

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Session XI: Non-nuclear waste

Assessment and classification of radioactivity in electrode coating and sludge from magnesium production

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Abstract: On request from HMN Demolishing Project, Institute for Energy Technology has assessed the levels of radioactivity in waste originating from the now decommissioned magnesium production facility at Herøya in southern Norway. A total of 418 electrodes and about 335 tonnes of sludge have been stored at Herøya.

A pilot study was initiated to assess the distribution and levels of radioactivity throughout the storage facility. Neutron activation analysis was used to determine the levels of ^{238}U and ^{232}Th in numerous samples. The feasibility of using hand-held monitors for the final assessment on site was tested.

Introduction

In 1999 Institute for Energy Technology (IFE) measured levels of radioactivity in samples of waste originating from the magnesium production facility at Herøya (Backe, 1999). The results showed that ^{238}U was the predominant radionuclide in the material. Low concentrations of ^{40}K, ^{210}Pb, ^{210}Po, ^{226,228}Ra, ^{232}Th and ^{228}Th were also detected, but the activity levels of these nuclides were significantly lower than the activity levels of ^{238}U.

After decommission of the magnesium production plant, classification and further treatment of the waste had to be considered, and in 2004 IFE was asked to assess the levels of radioactivity throughout the whole storage facility. The waste was divided in three categories; fixed coating on the electrodes, sludge (oxidised coating) from the electrodes and sludge from the galvanic baths. All sludge was stored in plastic barrels.

Uranium is present in both dolomite and seawater, both used as raw materials in the production process utilized at the magnesium production facility. The amount of ^{238}U in seawater is about 3 μg litre^{-1}, equivalent to 0.037 Bq litre^{-1} (Tsytsugina, 1973) and large volumes of seawater were needed for the production. The concentration in dolomite can vary from low concentrations up to a maximum value of about 1300 Bq kg^{-1} (Backe, 1999).

In 1999, the Norwegian Radiation Protection Authority (NRPA) specified an exemption level of 5 Bq g^{-1}for ^{238}U in the material. In addition, an exemption level of 5 Bq g^{-1} for ^{232}Th in the material was given prior to the work presented in this paper.

Pilot study

Samples were collected from 15 units (barrels or electrodes) of each category. To assess the homogeneity within one unit, 3 parallel samples were collected from 5 of the units. All samples were homogenized (grinded) and a smaller portion was analysed for ^{238}U by using neutron activation analysis (NAA). The results are shown in Figure 1. The remain-

ing part of each homogenized sample was packed in 100 ml plastic containers and measured using an alpha and beta sensitive contamination monitor (NE Electra with DP2 probe) for 30 seconds.

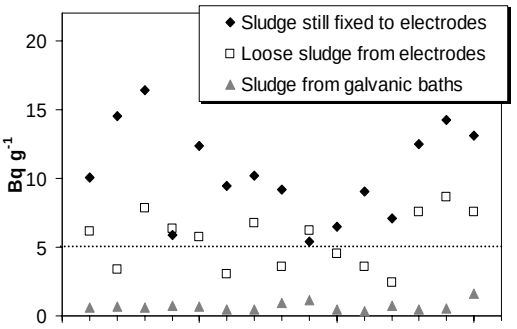

Figure 1. NAA results from the pilot study. The exemption level is indicated at 5 Bq g^{-1} (dotted line).

Parts of the samples were not possible to grind using available equipment. When measuring this fraction using the contamination monitor, a significant higher reading was observed for most of the samples. To decide whether this was due to an actual difference in activity concentration or other factors as self-absorption or geometry, some of the samples were attempted dissolved in HNO_3 and a portion of each solution analysed for ^{238}U. However, not all material was possible to dissolve in HNO_3. The results showed however that there was a significant difference in activity concentration in the two fractions.

Based on the NAA results, selected samples with varying activity concentrations were used as "calibration standards" for the field set-up, and the resulting calibration was used to calculate the amount of ^{238}U in the rest of the samples (both homogenized and non-homogenized fractions). These results were then compared with the NAA results (Figure 2). The same geometry and distance to the probe were used for all measurements.

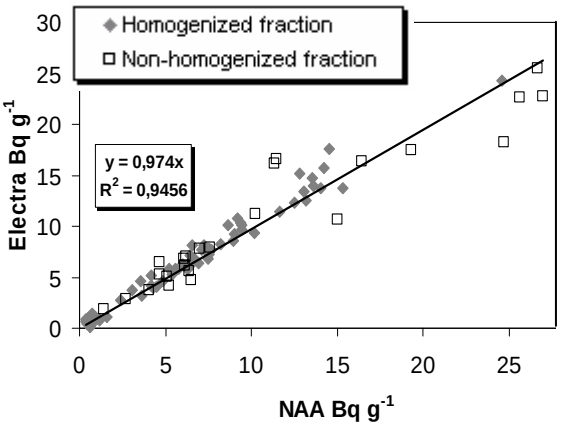

Figure 2. Comparison of NAA results and results obtained using the suggested field method.

A few samples were analysed for ^{232}Th to verify that it would be sufficient to focus all further work only on ^{238}U. The concentration of ^{238}U was between 4 and 12 times the concentration of ^{232}Th. Based on all results it was concluded that the field set-up was suitable for measuring ^{238}U in the material. This work is described in more detail by Strålberg (2004).

Sludge from galvanic baths

Based on the results from the pilot study, it was concluded that all sludge from the galvanic baths probably would contain concentrations of ^{238}U, and consequently also ^{232}Th, below the exemption levels. However, this had to be verified by field measurements. The homogeneity of the material varied. The standard deviation of the ^{238}U levels in 3 samples from the same barrel varied from 5 to 28% for different barrels. Because of the low activity level, this was considered not to be significant as compared with the exemption level of 5 Bq g^{-1}. However, to introduce a safety margin, an action level at 4 Bq g^{-1} was defined.

It was decided to collect one sample from every 10^{th} barrel and measure it using the method described above. If a sample from one barrel showed an activity concentration above the exemption level, five preceding and following barrels should be sampled. A field laboratory was established in the building where the waste material was stored. The set-

up (Figure 3) was calibrated using "standard" samples from the pilot study and sampling was carried out according to the procedure indicated above. The samples were put in containers identical to the ones used for calibration. Each sample was measured for 60 seconds and the background count rate was subtracted. The activity concentration of ^{238}U was then calculated according to the obtained calibration curve.

Figure 3. Field laboratory set-up at Herøya.

A total of 84 samples from 840 barrels were collected and measured. All samples showed activity levels of ^{238}U below the exemption level. The highest concentration measured was 3.4 Bq g^{-1}, average 1.2 Bq g^{-1} (standard deviation 0.7 Bq g^{-1}). This work is described in more detail by Strålberg and Mechkarska (2004a).

Sludge from electrodes

Based on the results from the pilot study, it was concluded that the activity levels of ^{238}U in sludge from the electrodes were either just below or well above the exemption level. The standard deviation of ^{238}U levels in 3 samples from the same barrel varied from 5 to 58%, average 25%. An action interval from 30% below to 30% above the exemption level (3.5-6.5 Bq g^{-1}) was therefore introduced. It was decided to collect one sample from each barrel and measure it using the same set-up as described above. The result should be compared with the action interval, and if the measured value was within this interval, two more samples should be collected and the average of these three results compared to the exemption level. Samples were collected from 110 barrels. The highest concentration of ^{238}U was 26.1 Bq g^{-1}, average 7.2 Bq g^{-1} (standard deviation 4.0 Bq g^{-1}). The activity was below the exemption level in 30 of the 110 barrels. This work is described in more detail by Strålberg and Mechkarska (2004b).

Electrodes

Figure 4. Classification of clean electrodes

All loose sludge was removed from the electrodes and collected in barrels (se above). Some of the coating was however impossible to remove and the electrodes had to be classified according to limits for fixed surface contamination. NRPA provided exemption limits for surface contamination of 1 Bq cm^{-2} for alpha- and 10 Bq cm^{-2} for beta emitters (average over 100 cm^2). As the work progressed the experience was that all electrodes with visible coating had to be classified as radioactive, and these were therefore set aside. Clean electrodes (Figure 4) were controlled against the limits from the NRPA using a contamination monitor (Electra with DP2 probe). Probe specific efficiencies for alpha- and beta radiation were used when calculating the surface activity. A total of 102 of 418 electrodes were free classified according to the provided limits. Over time (1-2 years) the coating will oxidize and become brittle. The remaining electrodes can then be cleaned and classified according to the exemption levels provided. This will significantly reduce the waste volum as the

activity only is present in the coating and not in the electrode material it self. This work is described in more detail by Strålberg and Mechkarska (2004b).

Discussion and conclusion

Similar field methods for analysing NORM have previously been used by IFE in decommissioning projects and classification of LSA scale from the oil and gas industry containing enhanced amounts of ^{226}Ra and ^{228}Ra. This was however the first time a similar method was used for ^{238}U. Because the industrial process leading to the formation of this material will disturb any possible radioactive equilibrium in the raw materials, it was necessary to measure ^{238}U directly, and not through its daughters, to determine the amount of the nuclide in the material. NAA proved to be a suitable choice for this, although the method requires the material to be fine powder or in solution. Not all material was possible to grind or dissolve in HNO_3. It was also necessary to leave the samples for 1-2 weeks after irradiation before the activity could be determined because of short-lived nuclides generated during irradiation.

The field method for analysing ^{238}U in sludge from magnesium production was suitable for measuring a large number of samples during a rather short time period. For activity levels above 2 Bq g^{-1}, the maximum deviation between the NAA analyses and the field method was about 35 %. The average deviation was -3 %. Only 6 samples showed a higher deviation than ±20 %. The main problem when conducting field measurements like this is the inhomogeneity of the material. It is highly recommended to carry out a pilot project to study this problem before the fieldwork is commenced. It will not always be appropriate to collect only one sample from each unit when the unit is large. However, time is also often a critical factor in projects like this, and it might therefore not be practical to collect several samples from several hundred units. Care should therefore be taken to establish sound action levels in order to take into account varying homogeneity of the material during field measurements.

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Problems with the management of non-nuclear radioactive waste in Sweden

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Abstract: There are shortcomings in the current system for the management of non-nuclear radioactive waste. Old disused sources i.e. from industry, research and hospitals are sometimes quite expensive to dispose of. The sources are then kept longer than needed and with time there is an increasing risk that they might end up as orphan. SSI has at present no legal rights to make this situation change.

Natural radioactivity may be accumulated in large volumes from industrial processes and will then require safe disposal in places yet to find.

The Government will have to consider the proposals that have been presented by a committee that has investigated the problems [1]. The legislation needs some changes and SSI needs new authorizations and resources to be able to reach the goal to have a functional and safe handling of all radioactive waste in Sweden. Decisions are also needed on how to implement the two EC-directives in connection with these questions (WEEE 2002/96/EG [2] and HASS 2003/122/EURATOM [3]). These directives are both supporting the goal and should be implemented and put into force during 2005.

This is a small essay with selected examples of unsolved problems with the management of radioactive waste from activities outside of the nuclear power production sector that we have to find practical and safe ways to handle.

Studsvik – “The one and only waste facility”

Once upon a time (1947) the business at Studsvik started as a state undertaking. From around 1995 the state has no ownership left and Studsvik AB is now a public limited company with about 8000 shareholders. There is in Sweden no alternative to Studsvik when it comes to waste management of radioactive material originating from outside the nuclear power cycle. It is therefore a dilemma whenever this monopoly company, on whatever reason, declines to take care of certain kinds of radioactive waste. Also when it is possible to take care of a special kind of waste in Studsvik the cost, which can often be high, can make the disposal impossible. These are some of the problems of today that we have to solve.

Smoke detectors

According to the EG Directive WEEE [2] all waste from disused smoke detectors from households has to be taken care of by a joint responsibility of the companies that sell them. This also includes the so-called historical waste originating from all detectors put into the market before August 13, 2005. The number of smoke detectors with a radioactive source currently in use in households in Sweden is estimated to be more than 10 millions. This is the extent of the historical waste. A majority, probably close to 100 percent, of smoke detectors produced for Sweden after this date August 13, will not include radioactive sources and will thus not add to the problems with radioactive waste disposal.

Never the less the producers of these new (optical) smoke detectors will have to pay for the disposal of the historical waste of old smoke detectors from the households.

The disused smoke detectors from households are taken care of in the same system that already exists for other disused electronic equipment from households. The waste can be disposed of at the municipal waste recycling station without any cost for the household. Such a system of collection of electric and electronic scrap has been in practice for four years in Sweden. The result is that we have already a collection of more than 100 000 smoke detectors in store waiting for further waste management. Money is needed for one thing.

One urgent problem is to find the price tag for the waste management. How much should the price of a smoke detector increase to cover the cost of the waste? The cost might be separated into two parts, one is to cover the pretreatment to take out the source from the detector, and one other is to cover the cost of final disposal of the source. The cost for the disassembling will easily be found by some practical tests but the other part might be difficult to solve. The only realistic repository to use for the sources of Am-241, with over 400 years of half life, will be a repository that in the future is going to be built by the Swedish Nuclear Fuel and Waste Management Company (SKB), for the long term nuclear waste. Therefore, the cost for final disposal can only be estimated and it is also uncertain when this repository will be finished and this implies a long period of intermediate storage with uncertain costs. Nevertheless, Studsvik has, though its newly incorporated division ISS under Stensand, offered a price of 25 SEK a piece, for an all inclusive waste management of smoke detectors.

Disused smoke detectors from use in hospitals, hotels, and other enterprises will further add to the problems with the ones from the households. The estimated number currently in use is less than three millions. This is the historical waste and also the total waste since all coming detectors that are produced for enterprises will not have radiation sources.

The producer's responsibility according to the WEEE-Directive differs for equipment delivered to enterprises compared to households. The responsibility is here reduced to a one-to-one exchange affair. That means that if a company buys a new equipment from a producer then the latter has to take care of a similar disused equipment without extra charge. Suppose then that we have a building equipped with hundreds of smoke detectors in a network connected to a fire-central and that this building is going to be demolished. Then there is no requirement for a producer to take care of the waste for free. The risk is that the company that owns this building might put the disused smoke detectors into the free waste stream intended for the households in order to save money in a simple way. It is not even clear if this is illegal, based on the present system of regulations for radiation protection. This is another question that has to be taken under consideration.

Source: Ref [1] and [2] and direct communication with the Smoke Detector Producer Group.

Contaminated filters from water works

Some places in Sweden have uranium in the drinking water in such amounts that it will have to be reduced by filtering. Natural occurrences of radium in drinking water might also be collected in the water filtering systems. The result is that some filters used in the water works for a long period of time will have to be treated as radioactive waste the day they are disused.

A yearly dose from drinking water of 1 mSv will be reached if the water contains 0.5 Bq or more of radium or 100 micrograms or more of uranium per liter drinking water.

From a toxic aspect the levels of uranium has to be even smaller. In USA the maximum permitted level is 30 micrograms of uranium per liter drinking water. Canada has 20 and WHO recommends a limit of 15 micrograms of uranium per liter drinking water. The WHO's recommendation equals 175 Bq of uranium per liter.

If the content is higher than these numbers some filtering is called for.

As an example the drinking water in Uppsala contains 30 micrograms of uranium per liter and in Köping it is 50.

The Swedish national food administration is planning to use the WHO's limit. This implies that large volumes of water with uranium have to be filtered and might thus generate waste with uranium that has to be handled as radioactive waste.

How and where this waste should be treated and deposited is for the moment not clear.

Source: SSI report 2004:14, A survey of natural radioactivity in drinking water [4] and direct communication with Swedish national food administration.

Colour pigment with uranium used at a porcelain factory

The Rörstrand porcelain factory has reported to SSI that there are three oil drums of nitrated uranium at the site. This substance was earlier used as a black color in the production of porcelain products but has now not been used for many ears. The radioactive waste is most likely depleted natural uranium.

It now remains for SSI to decide what the reasonable treatment of this waste should be.

Source: Request on waste management recommendation addressed to SSI.

Precision balance with radioactive sources

Balances from Mettler (types M5, B5 and B6) manufactured from the 1955 to 1960 were equipped with up to seven sources of Ra-226, each about 100 kBq. The purpose was to eliminate static electricity that would otherwise have influence on the readings. The problem arises when such a balance is taken out of use and has to be handled as waste. The information of the existing radiation sources is not easily found since no such signs are found on the balance itself. It is therefore a risk that these balances and similar ones from Satorius are treated as non-radioactive waste with following contamination later in the waste management chain of treatment.

Will it be possible to find all users and give them information of the problem and will this information still be at hand the day the balance is disused? There is no demand for a license in connection to the use and possession of such a balance.

Source: Request on waste management recommendation addressed to SSI.

Old cloths with radium

A family with children mowed to a house that earlier belonged to an ancestor and in the cellar they found three small cloths or pads with a sign indicating that they are prepared with radium for use in therapy. The mother called SSI and asked what to do. SSI made a quick check with Studsvik and it was clear that the cost for disposal would be in the re-

gion of € 1000. This is of course normally not affordable for a young family. All SSI could offer was to send someone to make measurements. The young mother answered that this should not be necessary. Hopefully we can come and collect this waste and similar cases without costly consequences for the holders.

Source: Request on waste management recommendation addressed to SSI.

What happened to the IKA-investigation?

The report from the investigation was delivered to the government in December 2003 [1]. The report presented a number of suggestions on how to solve the different problems in this area. It has also been submitted for comments according to the Swedish standard procedure and the comments from different interested parties were in most cases in agreement with the conclusions of the investigation.

After this only the most urgent questions have been handled. The government have addressed the two EG Directives that are involved; the WEEE Directive on waste electrical and electronic equipment that specially includes the smoke detectors [2] and the HASS Directive on the control of high-activity sealed sources and orphan sources [3]. Both directives are coming into force during 2005, the first already two weeks ago in August 13. At the time of writing SSI still has no information on where to find the funds to be able to take care of old waste that different holders now and then ask SSI for advice on how to get rid of.

SSI has suggested that a fund for restoration of industry polluted land areas should be opened also for taking care of radioactive waste that otherwise might end up polluting the environment on unexpected places. Hopefully this can be one way to get the necessary resources to handle some urgent cases.

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Establishing of a final repository in Denmark for LILW

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Abstract: The Danish Parliament has agreed to initiate the work to establish a 'Basis for Decision' concerning a Danish disposal facility for LILW. The principles for safety and environmental protection for a Danish repository are being prepared and are based on recommendation from international organizations. These principles are to set the frames for the rest of the process. It is generally difficult to describe the decision process in details. However, it is the intention to make the process as transparent as possible to the public, thereby allowing stakeholders to participate actively in the decision process. The type of repository should be based on types and amounts of waste as well as Danish geology.

Introduction

The Danish Parliament has in March 2003 agreed to the costs and the general decommissioning approach for all the nuclear facilities at Risø with the objective to decommissioning all nuclear facilities at Risø as soon as possible within a timeframe of 20 years. At the same time, the Parliament agreed to start the work to establish a 'Basis for Decision' on a Danish disposal facility for low and intermediate level waste. The 'Basis for Decision' is scheduled to be presented to the Danish Parliament in the fall 2005, and must describe how to proceed with the project of establishing a final repository.

Principles for safety and environmental protection

The fundamental principles for safety and environmental protection for a Danish repository is based on recommendations from international organizations, especially 'The Principles of Radioactive Waste Management', IAEA (1995), to ensure that the waste management will live up to the Joint Convention. These fundamental principles are to set the frames for the rest of the process. The following fundamental principles are recommended in the "Basis for Decision":

- **Protection of humans health and the environment**
  Radioactive waste shall be managed in such a way as to secure an acceptable level of protection for human health and the environment.

- **Protection beyond national borders**
  Radioactive waste shall be managed in such a way as to assure that possible effects on human health and the environment beyond national borders will be taken into account.

- **Protection of future generations**
  Radioactive waste shall be managed in such a way that predicted impacts on the health of future generations will not be greater than relevant levels of impact that are acceptable today. Furthermore, radioactive waste shall be managed in such a way as not to impose undue burdens on future generations.

Specific dose constraints
Specific dose constraints are also proposed in the "Basis for Decision". These dose constraints have been decided based on international guidelines, legal requirements as well as dose constrains used in other countries.

During operation the dose constrain is proposed to be 0.1 mSv/year which is equal to the dose constrain used in Denmark during operation of Risø and now during the decommissioning.

After closure, the dose constrain is proposed to be 0,01 mSv/year for likely events and 0,1 mSv/year for unlikely events. Precisely what is characterized as likely and unlikely events has not been established yet.

General thoughts and ideas regarding the technical solution of the repository

The type of repository is at the present moment not decided, and a long and democratic process is to be carried out before the decision is finally taken. The final choice will among others depend on adequate geological formations, type and quantities of radioactive waste as well as the question of reversibility.

Waste and capacity
The overall policy and practice for radioactive waste management have so far been to collect and store all Danish radioactive waste under safe and secure conditions at dedicated storage facilities at Risø. The stored radioactive waste comprises waste from the operation of the three research reactors and other nuclear facilities at Risø as well as radioactive waste from use of radioactive materials for medical, industrial and research purposes in Denmark

By far the largest part of the Danish waste is LILW-SL. However, a small amount is considered to be LILW-LL. Denmark does not have any HLW.

The final repository is expected to accommodate at least 5,000 m^3 conditioned waste, and it could be as much as 10.000 m^3 conditioned waste. The volume depends among others factors on what type of containers there are to be used and if it should accommodate for future waste.

Geology
Regarding geology, Denmark has sedimentary geology of primarily sand, clay and limestone. Furthermore, the water table is generally located relatively close to the surface.

The bedrock is located more than 500 m below surface (Except on the island of Bornholm).

Based on the IAEA recommendations, a repository should be located in impermeable geological settings.

Reversibility
There is an ongoing discussion regarding the matter of reversibility.

Type of repository
Several types of repository could be considered:

1) A near-surface repository. Potentially combined with a deep borehole for small amounts of long-lived waste. The near-surface repository can either be above or below the water table.

2) A deeper near surface repository located maybe 30-80 m below the surface.

3) A geological repository

Principles and ideas guiding the decision process of the project

The idea is to make the process of establishing a final repository a stepwise process where small steps are taken one at a time. Before each steps there should be general agreement about the step taken and it should be well documented.

Initially, all three types of repositories are considered a possibility and no locations in Denmark are ruled out as host. Then step by step the type of repository and the location are narrowed down based on sound arguments until a final decision can be made.

It is the intention to make the process as transparent to the public as possible, thereby allowing stakeholders to participate actively in the decision process. During the process there will be carried out several hearings and information meeting as well as publication of information material.

A leaflet has been prepared with information on the project. The leaflet was sent to all municipalities and a number of interested parties and NGO throughout the country. The leaflet is downloadable from the homepage of the Ministry for the Interior and Health. Following this, it is the intention to have one to two mini-seminars for interested parties. All working papers are accessible at the homepage of the Ministry for the Interior and Health, in order to allow interested parties to follow the process.

The future process

A figure of the tentative plan for the establishing of a repository is presented in figure 1. The process is presently in the beginning of Phase 1, so a long road is lying ahead.

If the "Basis for Decision" is accepted by the Parliament, the next step will most likely be to make feasibility studies of the potential repository types and a geological desk top survey of the Danish Geology. Based on this, a number of sites should be selected for intrusive investigation.

Figure 1. Tentative plan of phases in the process of establishing a final repository

	Planning		***Operative activities***			***Decay***	
Phase 1: Decision							
Phase 2: Site selection							
Phase 3: Establishing							
Phase 4: Operation							
Phase 5: Closure							
Phase 6: Monitoring							
Phase 7: Passiv phase							
Year (tentative)	2004			2050			2350

The exact method for selection of a final site is not decided yet. But after the intrusive investigations of a number of sites it is likely that a new document, being more specific

with regard to type and location, must be presented for the parliament for their acceptance.

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A screening calculation of radiological effect of airborne release from non-nuclear power facilities in Sweden

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Abstract: A simple Gaussian plume model was used to predict the concentration distribution of releases of elemental tritium into surrounding environment from two different sources. The Monte Carlo simulation approach was used to capture the effect of uncertainties caused by the input parameters. The simulation results indicated that the releases in general caused very low doses. However the ambiguity regarding the chemical form of tritium entails the greatest uncertainties in the calculations in comparison to estimated uncertainties in the input parameters. The dose conversion coefficient for tritiated water is four orders of magnitude greater than for elemental tritium, which makes the assumption of the amount of uptake of tritiated water in the lung critical for the calculations. Consequently the maximum doses may range from nSv to μSv per year. Therefore further studies are needed to achieve more realistic assumptions and to validate the model to be able to reduce the predicted uncertainties.

Introduction

To revise the existing regulations (SSI FS 1983:7) on handling of radioactive waste from non-nuclear power facilities, such as the hospitals and reasearch laboratories, SSI recently initiated a project, PULS, a Swedish abbreviation of limitation of release from hospitals and laboratories. In this project a modelling study to analyse the radiological consequences of airborne releases of tritium was performed in order to get a scientific basis for setting the release constraints.

Radionuclides released into the atmosphere are transported by the wind and dispersed by the turbulent diffusion and may ultimately lead to exposure of man. A range of models have been developed to address this advection-dispersion transport problem depending upon the simplifying assumptions made and boundary conditions imposed. The Gaussian Plume Diffusion Model (Slade, 1968) is widely used because of the relative simple calculation but the model is based on fundamental concepts of turbulent diffusion (Till and Meyer, 1983). The Gaussian plume model is recommended by NRPB (1979) for estimation of dispersion in the short and medium range, that is from 100 m to a few tens of kilometres from the source and it is still being used as a screening tool to evaluate the impact of individual plumes from point source releases (e.g. Smith et al., 2004). The Gaussian plume model is used in this study. The objective release sources are two pharmaceutical companies in south of Sweden (A and B), geographically closely located. Both companies release large amounts of tritiumgas from their laboratories when using Tritium Manifold Systems (Nilsson, 2005; Sjölin, 2005).

The parameters in the model most uncertain are the wind velocity and the standard deviation of the Gaussian distribution both in vertical and horizontal directions. The wind velocity can vary a factor of 8 during a day based on the wind data obtained from the meteorological observation station nearest to the source. The standard deviation of the Gaussian distribution is a function of distance from the release point and dispersion coefficients related to weather stability categories which in turn can vary with the whole range of the categories within a day (Smith et al., 2004). Commonly, one uses arithmetic average of parameter values in the model to estimate a “mean” output value from the model. However, expected output value of a function with parameter values related to certain distributions is not equal to the output value of the function with arithmetic average of parameters if the function is non-linear (Wörman and Cvetkovic, 1995). We use the Monte Carlo simulation approach to capture the effect of uncertainties caused by these parameters. A sensitive analysis is performed, which can guide us to reduce the uncertainties in the simulation in further studies.

Methods for screening calculations

Models: Taking into account for the reflection of plume from the ground and an inversion due to positive temperature gradients in the lower levels of the atmosphere, the formulation of the Gaussian model for an elevated release may be written as (NRPB, 1979):

$$C(x,y,z)=\frac{Q}{2\pi u_{10}\sigma_y\sigma_z}\exp\left(-\frac{y^2}{2\sigma_y^2}\right)F(h,z,A) \quad (1)$$

in which

$$F(h,z,A)=\exp\left(-\frac{(z-h)^2}{2\sigma_z^2}\right)+\exp\left(-\frac{(z+h)^2}{2\sigma_z^2}\right)+\exp\left(-\frac{(2A+z+h)^2}{2\sigma_z^2}\right)+\exp\left(-\frac{(2A+z-h)^2}{2\sigma_z^2}\right)$$
$$+\exp\left(-\frac{(2A-z+h)^2}{2\sigma_z^2}\right)+\exp\left(-\frac{(2A-z-h)^2}{2\sigma_z^2}\right) \quad (2)$$

where C is the air concentration [Bq/m^3] or its time integral [Bq s/m^3]; Q is release rate [Bq/s] or total amount released [Bq]; u_{10} is wind velocity at 10 m above the ground [m/s]; σ_y is standard deviation of the horizontal Gaussian distribution [m]; σ_z is standard deviation of the vertical Gaussian distribution [m]; x, y, z are co-ordinates for along the wind direction [m], cross-wind (horizontal) [m] and above ground (vertical) [m], respectively; h is effective release height [m] and A is the depth of the mixing layer (from the ground to the inversion) [m].

Experiments have been performed to derive empirical descriptions in estimation of standard deviation of Gaussian distribution under various weather stability categories (Till and Meyer, 1983). In the Jülich system, the descriptions were obtained at the sites with high surface roughness, which are more similar to the sites of this study. The standard deviation of vertical and horizontal Gaussian distributions in Jülich system are expressed as the following

$$\sigma_y(x)=p_y x^{q_y} \quad (3)$$
$$\sigma_z(x)=p_z x^{q_z} \quad (4)$$

where p_y, q_y, p_z and q_z are empirical coefficients.

From the concentration of tritium in the air, the inhalation dose due to airborne tritium release can be estimated by the equation below (Smith et al., 2004):

$$D_{inh}=\sum C_{a,i}\cdot I_{inh}\cdot T_i\cdot H_{inh} \quad (5)$$

where $C_{a,i}$ is concentration of radionuclide in the air [Bq/m^3]; I_{inh} is inhalation rate, [m^3/h]; T_i is exposure time [h/year], which is the release duration for each release; and H_{inh} is dose coefficient for inhalation [Sv/Bq]; i is the index for each release, i=1,2,….

Input data: The total amount of tritium discharged from site A was 425 GBq during 2004 and the distribution of each release and release duration are shown in Table 1. The total amount of discharges from site B was 25 GBq, with an average of 1850 MBq per release. The release occurred promptly from the stack, with a measured height of 17 m. Similar stack height was assumed for case A.

Since no meteorological measurements were performed at the release occasions the wind data was collected from the nearest observation station of Swedish Meteorological and Hydrological Institute (SMHI). The distribution of wind velocity measured at 10 m height with 3 hours interval during 2004 is shown in Fig. 1

The empirical coefficients used in Eqs. (3) and (4) can be found in Till and Meyer (1983) p. 2-23, Table 2.8. The mixing layer depth was chosen as 800 m as a normal weather situation at northern Europe (SSI, 2003). Inhalation rate, I_{inh}, was taken as 1 m^3/h (ICRP, 1974) and H_{inh} was 1.8 10^{-15} Sv/Bq for (adult) elemental tritium gas or 1,8 x 10^{-11} Sv/Bq (adult) for tritiated water (ICRP, 1996).

Methods for simulations: The wind data were used in the calculation with a total amount of 2928 data sets, which means 2928 realizations in the simulation. It is assumed that the standard deviations of the horizontal and vertical Gaussian distributions are independent of the wind velocity but highly correlated to each other. Corresponding to the wind velocities the 2928 realizations of the standard deviations of Gaussian distributions were generated by using Latin Hypercube sampling method based on an uniform distribution of various weather stability classes described by Eqs. (3) and (4). The reason to use the uniform distribution is that there is lack of information of the real distribution of the weather stability classes. Spearman Rank Correlation Coefficient method was used to perform the sensitive analysis because the method can deal with the non-linear relationship between input and output. All the simulations were performed in the MATLAB environment.

Results and discussion

Monte Carlo simulations based on Eqs. (1) and (2) for a prompt release of 1850 MBq tritium from site B were performed, which gave the mean, maximum and minimum concentration distribution versus transport distance along the central line 1 m above ground from the release point (Fig. 2). Summing the 10 releases occurring during 2004, the doses can be calculated according to Eq. (5) based on the concentration distribution obtained from the Monte Carlo simulations and the assumption that the exposure happens in a place where air flows continuously. The same wind direction was assumed for all the release occasions, which is thought to be conservative. The highest total dose occurs at about 80 m from the release point. The values of maximum, mean and minimum dose for elemental tritium for 2004 are $1.0 \cdot 10^{-11}$, $1.5 \cdot 10^{-12}$ and $5.8 \cdot 10^{-14}$ Sv/y, respectively.

Similar assumptions and simulations were performed for the releases of tritium from site A. Fig. 3 shows the tritium concentration caused by one release (2004-09-30) from site A along the central line in the wind direction 1 m above ground. The highest mean dose occurs at about 200 m from the release point. The values of maximum, mean and minimum dose for 2004 are $1.6 \cdot 10^{-10}$, $3.7 \cdot 10^{-11}$ and $6.6 \cdot 10^{-12}$ Sv/y, respectively. To locate the input uncertainties a sensitive analysis was performed. The sensitive analysis shows that the input variables σ_y and σ_z have the dominating influence on the output uncertainty. Both have a correlation coefficient of 0.8 on the output. Input variable u_{10} has a correlation coefficient of 0.4 on the output (Fig. 4).

The results from the screening calculation indicate that, although relatively high amounts of tritium are being released to air from one site, the estimated doses are far below 10 μSv per year. However, except the uncertainties caused by input data, that lead to the calculated dose varies with 2 orders of magnitude, arguments could be raised to the applied dose conversion coefficient. ICRP (1996) has estimated that about 0.01% of the tritium gas that is inhaled is assumed to be absorbed in the lung as tritiated water. Furthermore, McDonnell (2004) pointed out that "the magnitude of any inhalation dose will be very sensitive to the actual level of HTO and other chemical forms that might be present in a discharge of tritium gas". Hence, if we assume that 100% of the tritium released is absorbed as trititated water the calculated maximum dose from site A will be close to 10 μSv per year. The estimated ranges of dose clearly illustrate the significance of performing more realistic assessments. Therefore in order to demonstrate that the dose caused by the release of tritium in either case is below the environmental quality goal of 10 μSv per year further studies must be undertaken.

Table 1 Tritium release from site A during 2004

Date	Duration (hours)	Amount of release (GBq)
2004-01-20	5	36
2004-04-14	8	36
2004-04-15	5	44
2004-09-30	5	280
2004-10-06	5	29.3

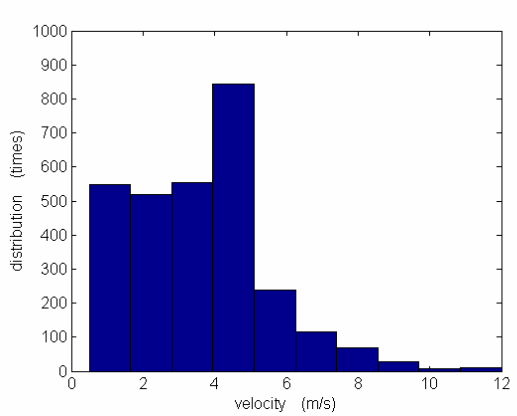

Fig. 1 Wind velocity distribution

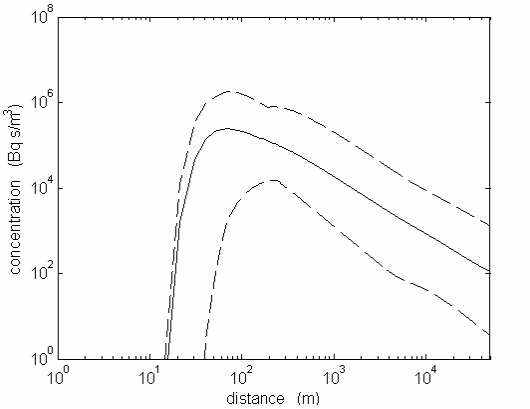

Fig. 2 Concentration vs. distance for one release (site B)

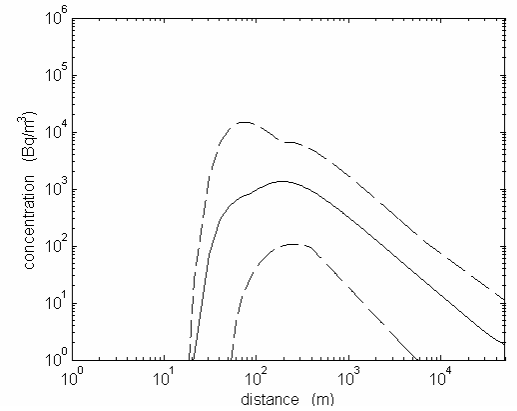

Fig. 3 Concentration vs. distance for the release on 2004-09-30 (Site A)

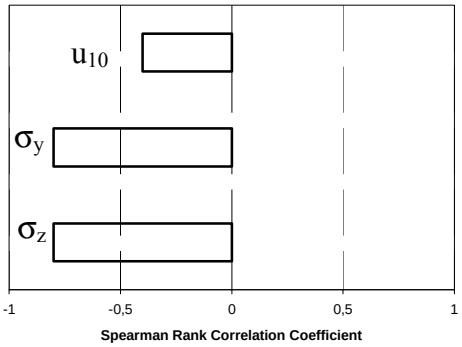

Fig. 4 Spearman rank correlation coefficients obtained from the sensitive analysis

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Controls of naturally occurring radioactive material (NORM) waste in Denmark

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Abstract: The Danish oil- and gas industry produces between 10 and 30 tons of low activity radioactive waste (NORM) per year. The principle radionuclide in the waste is Ra-226. Regulations in Denmark dictate a clearance level of 0.5 Bq/g for Ra-226. This means that parts of the waste produced from Danish oil- and gas fields have to be treated as radioactive waste. The Danish practices regarding treatment of NORM from the oil- and gas industry and include: 1) Storage of NORM waste. There are at the present no storage facilities for NORM waste in Denmark. As a consequence, the waste producers have to temporary store NORM, and 2) Re- injection of NORM offshore. NIHR has licensed re-injection of waste into abandoned production wells in the North Sea as an alternative to storage of NORM. Re-injection of waste offshore can minimize handling of the waste and eliminate the storage problem. However, this method requires approval by the relevant environmental authorities and compliance with international conventions.

NORM, Scale and Sludge

Scales with a content of naturally occurring radionuclides are a by-product of hydrocarbon extraction. Scales are found on offshore installations where injected water has mixed with water present in the formation. This causes formation of barium sulphate ($BaSO_4$) which precipitates as hard scale in pipes and production equipment or as a soft, water saturated sludge in separation tanks. Barium can be substituted by radium and all scale and sludge can thus be radioactive. The principle radionuclide found in scale and sludge in the Danish sector of the North Sea is Ra- 226 but all decay products of Ra-226 and Ra-228 can in theory be present.

The Danish Offshore Oil Production

There are 3 oil and gas companies, which operate in the Danish sector of the North Sea; Mærsk Oil and Gas, Amerada Hess and DONG (Dansk Olie og Naturgas). The total production of hydrocarbons was in 2004 22,614,000 m^3 of oil and 10,934 Million Nm^3 of gas. The number of oil and gas producing fields has been increasing since the first field started to produce in 1972 and until today where there are 19 producing fields (Energistyrelsen, 2005).

Offshore NORM Production

All oil and gas companies have fields that produce NORM. However, the amount and activity concentration vary a lot between the fields. The Danish oil- and gas industry produces between 10 and 30 tons of low activity radioactive waste per year. There is a tendency that water injection in oil- and gas fields enhances formation of radioactive scale in

the production equipment. Therefore, the amount of waste increases with the increased use of water injection in oilfields. It is normally the Ra-226 concentration in oil field waste that decides if the material is above the clearance level and thus needs to be treated as radioactive waste.

Licence for handling and disposal of NORM

Since 2002, Danish regulations regarding the control of NORM is given by the Exemption Order (Order no. 192, 2002). The Order sets exemption levels for naturally occurring radionuclides (Table 1). Exemption and clearance levels are the same for naturally occurring radionuclides.

Table 1. Exemptions/clearance levels for the principle naturally occurring radionuclides found in offshore NORM.

Radionuclide	Concentration (Bq/g)
Ra-226	0,5
Ra-228	1
Po-210	5
Pb-210	5

All oil and gas companies that produce NORM with activity concentrations above the exemption level must have a licence from NIRH to handle, store or dispose NORM. The licence is given under the following conditions:

- There must be appointed a radiation protection officer with the appropriate qualifications and knowledge of Danish regulations regarding NORM.
- There must be relevant instruments for measurement of ionizing radiation and radiation protection equipment available.
- There must be written working procedures available for all work concerning NORM. The working procedures must include at least information regarding, safe handling of NORM, dose control, measurements of dose rate and surface activity, and procedures for sampling of NORM for laboratory analyses.
- A annual report with all measurements and analyses of NORM must be submitted to NIRH.

Licenses for disposal and storage of NORM are given on a case to case basis.

Disposal options

NORM produced from offshore activity has so far been handled of in two ways: 1) storage and 2) re-injection into abandoned production wells in the North Sea.

Storage: There is no recognized installation for disposal of NORM in Denmark. As a consequence, the waste producers have to temporarily store NORM. The waste is normally stored in plastic barrels in a controlled area. Although, the interim storage sites present no immediate capacity problems, there is a pressure from the industry for a more permanent solution to the disposal problem. Furthermore, it must be expected that future decommissioning of the oldest rigs in the Danish North Sea sector will generate large amounts of NORM. This has been seen in Norway (Strålberg et al., 2003).

Re-injection of NORM offshore: NIHR has in one case licensed re-injection of waste into abandoned production wells in the North Sea as an alternative to storage of NORM. Re-injection of waste offshore can minimize handling of the waste and eliminate the storage problem. This method requires approval by the relevant environmental authorities and scrutiny of international conventions. The conventions that deal with the marine environment in the North Sea include The London Convention and the OSPAR Convention. These conventions are administrated by the Danish Environmental Protection Agency (DEPA). NIRH is therefore not able to issue a license that involve release of NORM to the marine environment or the subsurface seafloor without approval from the DEPA. DEPA has among other things ruled that waste can only be re-injected if this is done on the same offshore installation where it is produced. It is thus not possible to move waste from one platform to another or from the shore to the offshore for re-injection. This, of cause, limits the types and amount of waste that is suitable for re-injection.

Conclusion

A recognized installation for disposal of NORM in Denmark will meet the need from the industry for a permanent solution to the waste problem. However, re-injection of waste into abandoned wells is another solution that reduces the amount of waste that needs to go onshore for disposal. Although, this method can not be used for all scenarios and all types of NORM, it should be considered as often as possible.

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Radiation doses to staff at sewage plants from radioactive waste in nuclear medicine in Sweden

H. Jönsson[1], S. Sundell-Bergman[1], R Avila[2]

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Abstract: In 2003 the Swedish Radiation Protection Authority started to revise the regulations on releases of radioactive waste from hospitals and laboratories. In this framework a project was undertaken to assess the radiological consequences of radionuclide releases from hospitals in Sweden. The most frequently used radionuclides are Tc-99m and for therapy I-131. The results of the modelling show that the doses for sewage workers become high using conservative assumptions and there is a need too apply more realistic data in the models. Measurements are therefore undertaken at some locations for validating the models.

Administered activity to patients in Sweden (GBq) per year

	2000	**2001**	**2002**	**2003**
C-14	0.25	0.25	0.18	0.20
I-131	2100	1700	1900	2000
P-32	80	70	60	60
Sm-153	420	360	540	730
Sr-89	60	30	30	20
Tc-99m	34200	32600	33600	34300

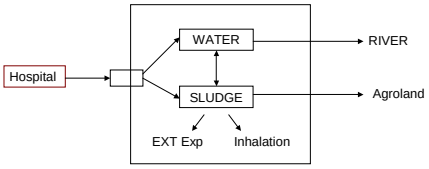
Our approach

- Review of sewage treatment and disposal
- Screening study based on generic models (*IAEA Safety Report No.19*)
- Data collection
- Development of dynamic model (LUCIA)
- Detailed studies

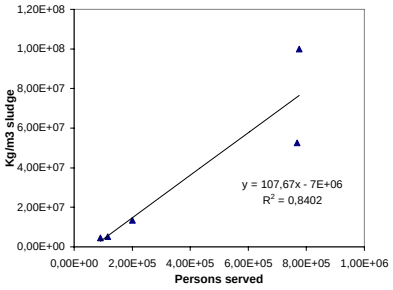
Screening study

- IAEA conservative methodology
- Assume no retardation from the hospital to the sewage plant.
  - *Case 1 - Assume no retention at the sewage plant.*
  - *Case 2- Assume full retention in the sludge of the sewage plant*

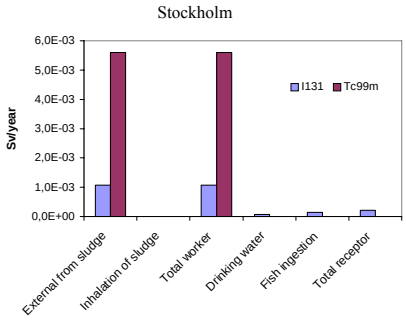
Screening model



Sludge production



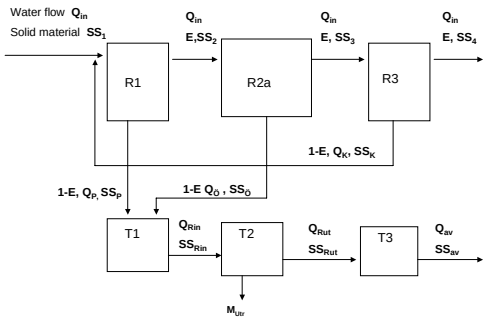
Conservative dose assessments



Conclusion

- Doses to public < < 10 mikroSv;
- Neglible impact from the radioactive waste;
- Doses to workers at sewage plant >10 mikroSv;
- Not neglible and need for more realistic assumptions;

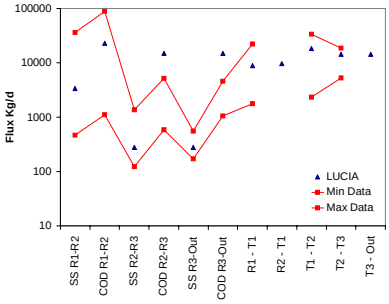
Detailed model



Doses to worker at SWP

	Release Bq/y	Screening Sv/y	SMART Sv/y	LUCIA Sv/y
I-131	7.8E+10 4.1E+11	8.0E-4 1.1E-3	7.3E-6 3.8E-5	6.5E-6 3.4E-5
Tc-99m	9.9E+11 6.6E+12	3.3E-3 5.6E-3	2.1E-7 1.4E-6	3.7E-8 2.5E-7

Test of LUCIA model



Uncertainties in dose assessments

- Lack of data from behaviour of radionuclides in sewage systems;
- Partition of activity between sludge and water in sewage systems;
- Unknown geometry at exposure station

Action plan

- Validation of LUCIA model
- Develop a more generic model applicable for sewage systems in Sweden;
- Dose calculations for releases of radionuclides from hospitals in Sweden;
- Develop generic release factors for unit release (Sv/Bq)

Regulations on radioactive waste generated by non-nuclear activities

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Abstract: The regulations concerning non-nuclear radioactive waste in Sweden is dated 1983. We are now in the process of rewriting them since a lot has happened in the field of radiation protection according to national and international regulations, agreements and recommendations. New areas in the regulations will for example be risk assessment and quality assurance programs.

Current regulations (SSI FS 1983:7)

- Applicable to liquid discharges and solid waste;
- Risks based on the ALI – concept *(ICRP 1977);*
- Radioactive waste generated by patients excluded;
- No reporting demands;

New requirements

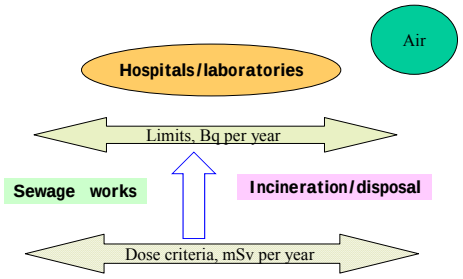
- Radiation protection law from 1988
- BSS (EU) 1996
- Environmental Code came into force on 1 January 1999
- International recommendations (*e.g. IAEA Safety Guide 2002)*
- Environmental Quality Standards

Radioactive waste management policy

- Radioactive wastes are not unnecessarily created
- Aim to safeguard the interests of existing and future generations and the wider environment
- Commands public confidence and takes account of costs

Radiological protection criteria for public exposure

- Dose limits , currently 1 mSv per year;
- Site and source contraints (e.g. 100 µSv per year at nuclear powers sites);
- Optimization (ICRP) in handling of wastes;



Approach

- Not practicable to assess doses to each individual member of the public- ”Critical group dose” concept;
- Include all relevant exposure pathways;
- Liquid discharges: two separate groups – sewage workers and public;
- Workers at sewage plants are treated as members of the public (no direct tangible benefits)

Dose calculations

- Dose assessments require assumptions;
- BSS (EU) demand realistic assumptions;
- Dose source/site constraint of 100 µSv per year
- Doses above 10 µSv are of regulatory concern;

Staged approach (1)

Simple and cautious assessments — 10 microSv/year

Detailed dose assessment (LUCIA) — 100 microSv/year

Monitoring (LUCIA) — 1 milliSv/year

Staged approach (2)

- Reference values for annual discharges for particular radionuclides based on critical group dose 10 mikroSv per year (**prospective**);
- Dose per unit release factors based on generic assumptions (100 mikroSv per year) (**prospective**);
- Monitoring of sludge in sewage plants (realistic) (concentrations based on 100 mikroSv per year) (**retrospective**)

Coming regulations…..

- Generic
- Waste characterised as releases to air and water, disposal of hazardous waste (incineration ashes);
- Criteria based on staged approach
- Waste generated by patients included
- Quality assurances
- Monitoring
- Reporting

Work plan

- Development of generic model for liquid discharges
- Dispersion calculations for discharges to air
- Review of scenarios, disposal of solid waste
- Calculation of reference values

Safe management of non-nuclear radioactive waste

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Abstract: In May 2002, the Swedish Government set up a non-standing committee for the management of radioactive waste unrelated to nuclear technology i.e. outside the nuclear fuel cycle – in this report called non-nuclear radioactive waste. The objective was to elaborate proposals for a national system for the management of all types of non-nuclear radioactive wastes with special consideration of inter alia the polluter pays principle and the responsibility of the producers. The author was principal secretary in the Committee. The proposals from the Committee was delivered to the Government by December 3rd, 2003 [1].

Funds for future costs for the management and final storage of waste from nuclear power are collected in a state-governed funding system. The power sector pays a flat fee per kilowatt-hour nuclear power. For non-nuclear radioactive waste, however, there are no means today to secure the funding. If a company goes bankrupt and leaves radioactive waste behind it might be up to the taxpayers to pay for its safe management. This is because the holder of the waste is responsible for its disposal. The costs appear at the time of disposal and it is usually the last owner/holder of a radioactive product that has to pay. Sometimes the costs come as a surprise and the owner might not have the money available. Thus the waste might be kept longer than warranted or end up as orphan waste.

To solve this dilemma and other weaknesses in the Swedish system the Committee proposes a funding system paralleling the system for nuclear waste. The cost for the waste should be paid up front, i.e. when a customer buys a product using a radioactive source the cost for the future waste management should be included in the price. In this way the consumer will not have to pay for this the day he disposes of the product by returning it to the original producer or leaving it to some waste treatment organization. It should be the responsibility of the producer (manufacturer, importer) to guarantee the funding for the handling of waste by making advance payments to the state fund.

PROPOSALS FROM THE COMMITTEE

The Committee's proposal for Product Waste:

All producers of products, with radioactive substances as components or products in the form of radiation sources, must be imposed with a producer responsibility for the waste that arises when the product or the radioactive component is discarded or scrapped (Product Waste). By producer, is meant the one that produces, imports into the country or releases products on the market that give rise to radioactive waste. This responsibility must include an obligation for taking back the products after completion of use and for being in charge of the final handling of the waste.

Producer responsibility should include a fee that would go to a special state fund for meeting the total costs of handling any arising radioactive waste, including final disposal if it is dealt with in Sweden.

When it comes to waste to be taken back by a supplier for final disposal in the country of origin, a bank guarantee or the equivalent should be insisted on.

The fee must meet a reasonable share of the total costs of the historic waste for which no financial producer responsibility was previously in force.

A State fund should be established for collecting and administering the fees that producers pay for the handling of non-nuclear radioactive waste (The IKA Fund), together with making payments under the authorization of SSI. The Fund should be tied in with the existing Nuclear Waste Fund and share the same administrative body.

SSI should be given a wider role when it comes to producer responsibility

- for checking authorization and keeping a register of producers
- for establishing and debiting fees
- for receiving compensation demands and approving disbursements
- for issuing instructions concerning handling and financing of IKA waste
- for checking and monitoring the waste handling system

A special advisory committee with representatives from the side of the producers and authorities should be attached to SSI for following up and scrutinizing SSI activities within this area. The committee should also be able to make decisions on recommendations concerning the activities.

The new system of producer responsibility for non-nuclear radioactive waste should come into effect on August 13th, 2005, concurrently with the new regulations that intend to implement the EC Directive 2002/96/EG [2], concerning waste from electrical or electronic products.

The Committee's proposal for Industrial Waste:

Radioactive Industrial Waste, i.e. radioactive waste that arises through enrichment of naturally occurring radioactivity or through the handling of biomass containing fall-out from nuclear weaponry tests or nuclear technology accidents, e.g. combustion of bio fuels, must be handled according to the Environmental Act (1998:808) [3] regulations, regarding ecologically harmful activities.

Operations that can be thought of as giving rise to radioactive Industrial Waste should, in the first place, be tested for authorization, according to the Environmental Act, and this must be preceded by a description of the environmental consequences regarding radiation risks and the genesis of Industrial Waste.

Operations that give rise to Industrial Waste but which have not been tested for authorization, according to the Environmental Act, must be regulated through the application of the Radiation Protection Legislation.

It must be possible for permits for ecologically harmful operations to include conditions about preventive measures that aim at avoiding or restricting radioactive waste arising, as well as conditions for how waste is to be managed and taken care of or to include a delegating to the supervisory authority to issue further regulations about the management of the waste.

In cases where the Environmental Court or a County Administrative Board issues conditions concerning the taking care of radioactive Industrial Waste that involves demands for financial security, an enterprising party should pay a fee to the IKA Fund to guarantee

that the waste can be taken care of correctly on the discontinuation of the operations. This is to be regulated in the legislation concerning the setting up of the IKA Fund.

The radiation protection legislation be altered, so that Industrial Waste is covered by the provisions of the legislation, concerning responsibility for radioactive waste through §13 of the Radiation Protection Act [4] being made applicable to Industrial Waste, by the concept of activities involving radiation being expanded by an additional paragraph in §5 of the same legislation.

SSI should

- where applicable, issue general instructions for the management and final disposal of certain Industrial Waste, with the support of the Radiation Protection Act, in accordance with the above mentioned amendments.
- be granted the right to be able to request a reappraisal of operations that give rise to radioactive Industrial Waste, according to Ch. 24 in the Environmental Act, and to be identified as the supervisory authority in the regulations (1998:900) [5] regarding supervision.
- be given the mission of informing about problems with radioactive Industrial Waste in the respective trade journals and to publish general advice on the management of such waste.
- confer with the Swedish National Environmental Protection Authority for being responsible for the supervisory and regulatory committee. SSI ought to be included in this committee.

The Committee's proposal for Other Waste:

Those, liable for payment to the IKA Fund, deposit a proportionate contribution to the Fund, in order to meet the costs for dealing with any other waste, where it is not possible to establish a legally responsible party for the waste.

Within their present inventory program and yearly suggestions concerning measures to decontaminate contaminated land and deal with hazardous waste in old industrial plants, the County Administrative Boards should be given the mission to include radioactive waste and radioactive contaminated land. SSI should be given the mission to assist the County Administrative Boards and the Swedish National Environmental Protection Authority with information and expert knowledge in this work. The appropriation for decontaminating and dealing with hazardous waste should, where appropriate, also be available for measures to decontaminate the radioactive waste.

Operational Waste

In addition to these three categories of waste proposed by the Committee, SSI has after its review of the proposals from the Committee, suggested an additional waste category: Operational Waste.

Operational Waste is radioactive waste that arises from non-nuclear activities with radiation such as protective cloths and disused laboratory equipment that has been contaminated during the work.

SSI has the authority to issue regulations for work with radiation and it is the operators that will have to take the responsibility to handle the waste generated in the operations.

And as proposed by the Committee those operating activities with radiation, which give rise to contaminated matter or induced radiation, must be able to be charged with paying the fee to the IKA Fund for dealing with the waste.

The Committee's proposal for Clearance:

SSI should be given the opportunity to make decisions about general exceptions to the Radiation Protection Act, when this is well founded, from the point of view of radiation protection and when it concerns the demands for an effective management of non-nuclear radioactive waste; a so called *clearance*.

Amendments should be made to the Radiation Protection regulations (1988:293) [6] to achieve this goal.

The Committee's proposal for Final Disposal:

At the earliest possible opportunity, the State should commence negotiations with the Swedish Nuclear Fuel and Waste Management Company (SKB), regarding a general agreement for the final disposal of non-nuclear radioactive waste, based on the grounds that the Committee discussed with SKB during the investigation. These imply, among other things, that SKB undertakes the final disposal of all IKA that requires such final repositories as in any of its existing or future plants and will do this for, principally, cost price. Furthermore they imply that SKB must be able to receive compensation also for actions taken for the adaptation of the storages and for the reappraisal of the licensing conditions for the repositories when necessary.

The Committee's assessment of Studsvik AB (Ltd.) in the Waste System:

The negative effects from Studsvik AB monopoly position on the market for services that concern IKA are going to substantially decrease, if the Committee's other proposals for a national system for IKA are realized. The grounds for Studsvik AB, and any other possible players on the market, for not handling certain kinds of radioactive waste, disappear in part or completely. The Committee therefore concludes that there is presently no need for any duty, regulated by law, to take care of IKA. Nevertheless, the Government ought to closely follow the development.

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6. SFS 1988:293. Strålskyddsförordning

Automated radiation monitoring at the Russian shipyards Atomflot and Polyarninski

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Abstract: The increased rate of decommissioning and dismantling of Russian nuclear submarines has created the need for improved radioactive waste management and radiation monitoring in Northwest Russia. The Arctic Military Environmental Cooperation (AMEC) programme addresses these needs. AMEC is a cooperative effort between military establishments of the Russian Federation, United States, Norway and the United Kingdom to reduce potential environmental threats from military installations and activities in the Arctic and enhancing the environmental security in the region. Results from the AMEC Nuclear Safety project AMEC 1.5-1 are presented.

The goal of this project is to enhance the ability of the Russian Navy to effectively and safely perform radioecological monitoring at selected facilities for dismantling of nuclear submarines and handling and disposition of spent nuclear fuel and radioactive waste. This has been accomplished by development of an automated and centralised radiological surveillance system based on the Norwegian software package PICASSO. The system has successfully been installed and is in regular operation at Atomflot. The second installation of the PICASSO system is at the Naval shipyard FSUE 10 SRZ in Polyarny, northwest of Murmansk. The installation was initiated in October 2004 and will be completed in the fall of 2005.

Introduction

The Arctic Military Environmental Cooperation (AMEC) was established to provide a forum for Norway, Russia, and the United States to work together in addressing military-related environmental problems in the Arctic. In September 1996, the Norwegian Minister of Defence (NOR MOD), the Russian Federation Minister of Defence (RF MOD), and the U.S. Secretary of the Department of Defense (U.S. DOD) signed a historic Declaration calling for cooperation among the parties to jointly address these environmental concerns. In June 2003, the United Kingdom – UK, joined AMEC[1]. The primary objectives of the AMEC Program are to: 1) share information on the impacts of military activities on the arctic environment, 2) develop cooperative relationships among military personnel in the participating countries, and 3) sponsor technical projects that assess the environmental impacts of military activities in the arctic and develop action plans and technologies for managing such impacts.

[1] The UK was invited to participate as full members, but decided to engage only in new projects started after June 2003, thus the UK did not participate in Project 1.5-1.

The goals of the AMEC Project 1.5-1 (Radiation Control at Facilities: Application of the PICASSO System) are to enhance and improve the technical means of the Russian Navy for measuring and controlling radiation exposure of personnel, the local population and the environment at sites involved in decommissioning and dismantling of nuclear submarines and handling and disposition of spent nuclear fuel (SNF) and liquid radioactive waste (LRW). This has been accomplished by the development, demonstration and installation of an automated centralized radiological monitoring system based on the Norwegian software package PICASSO at the Federal State Unitary Enterprise (FSUE) Atomflot, in Murmansk, Russia.

PICASSO-AMEC environmental monitoring system

The software package Picasso, developed by the Institute for Energy Technology (IFE), OECD Halden Reactor Project, in Norway, is a data presentation and visualization software which is well suited when large amounts of data are to be stored, transferred to a user interface and presented graphically in real-time in a user-friendly and flexible manner. More information on the Picasso software and its applications can be found at the website http://www.ife.no/picasso.

The Picasso software is used at IFE's research reactor in Halden to monitor radiation parameters in the reactor hall, the storage for SNF and experimental parameters from the reactor core. This application was demonstrated to Russian Naval officers in September 1998. Subsequently, a trilateral decision was made to initiate a project for application of Picasso at Russian facilities that handle military-related SNF and radioactive waste. The project was approved in February 1999. IFE programmers developed a prototype system for presentation of radioecological data, called the Picasso-Environmental Monitoring System. Russian Naval officers and programmers from the Nuclear Safety Institute of the Russian Academy of Sciences (IBRAE) received training at IFE, and the software was transferred from Norway to Russia in September 1999. Russian programmers and technical experts at IBRAE adapted the software to the Russian language and modified the system for use at naval bases. An operating model of a measuring unit, a working model, was developed, which includes terrestrial and underwater gamma detectors, smart controllers, radio-modems for off-site transmission of data, software for data acquisition and processing coupled with the Picasso-Environmental Monitoring System. More details about the earlier stages of this project have been presented by Endregard *et al*. (2002) and Pomerville *et al*. (2003).

Radiation monitoring at FSUE Atomflot

FSUE Atomflot, Murmansk, is the service base for the Russian Federation's nuclear powered icebreaker fleet. It is involved in preparing SNF for transportation by rail to Mayak, and receiving, processing and storing of liquid and solid radioactive wastes. An interim storage pad for 19 casks containing naval SNF, was commissioned at this site in November 2003 under the framework of AMEC. The liquid radioactive waste (LRW) treatment facility at this site has recently been modernized and upgraded by a US, Norwegian and Russian project, the Murmansk Initiative. Upon full commissioning, this facility will treat 5000 m^3 per year of LRW. These potential radiation hazardous activities call for corresponding measures for radiation control.

Since the existing radiation monitoring system at FSUE Atomflot had passed its useful operating lifetime, the AMEC program took on the task to install an automated centralised radiation monitoring system based on the PICASSO software. This is the first installation of the PICASSO-AMEC system. The construction work at FSUE Atomflot began in June 2003. Start-up and adjustment was carried out in the period 1 September to 16 September 2003. The six months trial operation was completed on 26 March 2004.

The installation of the PICASSO system includes terrestrial gross gamma sensors at the SNF pad, the location for SNF reloading and off-loading, and the LRW treatment facility, aerosol detectors in the ventilation exhaust pipes and a submersible water activity sensor in the sewage discharge pipe. The system provides remote stand-alone and continuous radiation monitoring with presentation of the data in real-time with the option of comparison with historical data. Alarm limits are defined. Both audio and visual alarms are given to the operators in the Radiation Safety Department and to key personnel in the administration building. An important advantage of the system is that it can easily be expanded with additional sensor types and locations at a later stage, as is being demonstrated by the PICASSO installation at FSUE 10 SRZ.

The installation at FSUE Atomflot includes:

- Dose rate detectors (BDMG-08R-03/04/05 from OAO "Pyatigorsky plant Impulse") - Eight channels for gross gamma measurement (Geiger muller); at the storage facility for SNF, the location for re-loading of SNF, and on the administration building. Sensitivity: approx. 0.1 μGy/h – 10 Gy/h.
- Water-monitoring detector (BDZhG-08C from SNIIP, Moscow) - One NaI detector for water radioactivity measurement in the sewage discharge pipe. Sensitivity (^{137}Cs): approx. 0.4 Bq/l.
- Air monitoring detectors (PVS-01 from MGP "Doza") - Two GM-counters monitoring aerosol activity on filters; at the location for re-loading of SNF and on the administration building. Sensitivity (^{137}Cs): approx. 0.1 μGy/h – 10 Gy/h.
- Aerosol detectors (UDA-1B from MGP "Doza") - Three silicium detectors for α- and β-aerosols control in the air in the ventilation system of LRW treatment facility. Alpha sensitivity: 10^{-2} -10^{5} Bq/m^3. Beta sensitivity: 10^{-1} -10^{6} Bq/m^3 (Sr(Y)-90).
- Weather station (MK-15 from NPO "Typhoon") - One meteorological station on the administration building that measures temperature, wind speed and direction, humidity and atmospheric pressure.

Radiation monitoring at FSUE 10 SRZ (Polyarninski)

The Polyarninski Shipyard is situated in Polyarny, a town of 25,000 inhabitants, north of Murmansk in the Kola Bay. The shipyard carries out maintenance work on laid up submarines as well as service of nuclear submarines that remain in service and has necessary equipment for refueling and dismantling naval reactors. Solid radioactive waste at the shipyard is placed in containers and stored in an open pad area, which is full. Liquid radioactive waste is stored in floating tanks at the quay. The shipyard dismantles first genera-

tion nuclear submarines. AMEC has taken on the effort to establish an integrated radioactive waste management complex (PPP-RAO) at this shipyard. The elements include the Picasso system for radiation monitoring, the mobile pre-treatment facility for solid radioactive waste, hydraulic metal cutting tools, containers for tansport and storage of solid waste, the mobile treatment facility for liquid radioactive waste, and a waste storage facility. This complex is described in more detail by Griffith *et al*. (2003).

The current radiological control system at this site is primarily limited to field measurements with hand-held detectors. The new system design includes dose rate detectors (GM) at the open pad for interim storage of solid radioactive waste, at the piers where submarines are laid up awaiting dismantling, at the floating tanks with liquid radioactive waste, at the floating docks where submarines are dismantled, at the entrance gate, and at the Radiation Safety Department building. Dose rate detectors will also be installed around the mobile pre-treatment facility (MPF) comprising a part of PPP-RAO. One aerosol detector for measurement of α- and β-emitters is to be installed in the exaust pipe from MPF. The submersible detector for water radioactivity (NaI) is planned in the sewage discharge pipe from the PPP-RAO installation. This position is beneficial for the integral assessment of the shipyard's impact on the water environment. In addition the system will contain a mobile gross gamma detector and a metrological station. In total 11 stationary gamma doserate detectors, one mobile detector, one underwater detector, one aerosol detector and one metrological station will be connected to Picasso. The installation was initiated in October 2004, and will be completed in fall 2005.

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Session XII: NKS (Nordic Nuclear Safety Research)

Nordic cooperation in an NKS perspective

T. Bennerstedt

NKS

Nordic Cooperation in an NKS Perspective

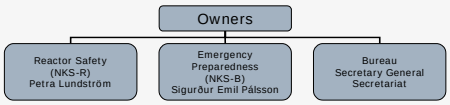
NSFS Conference, Rättvik
August 27 – 31, 2005

Torkel Bennerstedt
Secretary General, NKS
www.nks.org

Nordic Fora

- Nordic – Baltic Summits
- Nordic Summits
- NEP Group
- NSFS
- NKS
- Other groups, seminars etc.

Organizational Chart



Additional Funding

- Finland:
  Fortum
  TVO
- Norway:
  IFE Halden
- Sweden:
  Forsmark
  OKG
  Ringhals
  KSU
- Other national contributions

Why Nordic Cooperation?

- Common set of cultural values
- Shared historical background
- Cost-effectiveness
- Added international strength
- Complements EU, IAEA, OECD/NEA

Some NKS Objectives

- **Improved nuclear safety**
  (incl. rad.prot. and em.prep.)
- **Increased competence**
  and in-depth knowledge in these fields
- **Dissemination of information**
  in these matters
- **Networking**

Owners and Main Financiers

DK	Emergency Management Agency
FI	Ministry of Industry and Trade
IS	Islandic Radiation Protection Institute
NO	No. Radiation Protection Authority
SE	Sw. Nuclear Power Inspectorate
	Sw. Radiation Protection Authority

Economy DKK/yr.

Revenues	
Owners	7,0 millon
Others	0,6 million
Expenses	
NKS-R	3,2 million
NKS-B	3,2 million
Coordination	1,2 million
In-kind contr.	6,5 million

Program Areas

- **Reactor Safety**
  incl. decommissioning and radwaste
- **Emergency Preparedness**
  incl. radioecology and emergency preparedness related information and communication issues

Criteria for NKS Activities (1)

- **Nordic added value**, incl.
  - networks, groups, fora
  - competence enhancement and sharing
  - young generation of researchers
- Innovative scientific or technical work of high standard
- Holistic approach, transparency, openness
- Cost-effectiveness
- Certain amount of competition for funds

Criteria for NKS Activities (2)

- Clear goals that can easily be evaluated
- Net benefit to owners and end users
- Possibility to support PhD and MSc students
- Dissemination of results through, e.g.,
  - seminars
  - reports, articles, handbooks, manuals
  - CDs, webpages, newsletters
  - educational and info material

Criteria for NKS Activities (3)

- Coordination with work within
  - EU
  - IAEA
  - OECD/NEA
  - Nordic-Baltic cooperation programs
  - etc.

Future Issues

- Development (e.g., Olkiluoto 3)
- Decommissioning
- Radwaste
- Increased efficiency
- Cooperation
- Networking
- New media
- Suggestions are welcome!

How to Join NKS Work

- Join ongoing work by contacting the program manager (PM); or:
- Suggest new activity to PM
  - who evaluates the proposal
  - and presents it to the NKS Board
  - which might OK the proposal
  - and decides on budget, work plan etc.
- Welcome to join the NKS family!

Contact Persons

- NKS-R: Petra Lundström
- NKS-B: Sigurður Emil Pálsson
- Secretatiat: Annette Lemmens, Finn Physant
- Secretary General: Torkel Bennerstedt

- See www.nks.org
- Subscribe to NewsLetters & NewsFlashes

NKS-R: Nordic co-operation in reactor safety

P. Lundström

NKS-R

Abstract: The NKS-R program is structured around two main themes:

1. Development & Validation (DELI) of assessment methods and new technology. This theme covers the challenges related to the plant safety assessment and the introduction of new technology into the plants.
2. Management & Organisation (MANGAN) of safety and quality assurance. This theme covers the challenges related to the implementation and assessment of effective safety and quality management, and to the human performance in different situations.

Research activities are selected based on a Call for Proposal (CFP) proposal. The current program (including activities based on the CFP of autumn 2004) covers the following research areas:

- Thermal-hydraulics and severe accidents
- Risk analysis
- Organisational issues, safety culture
- Automation and control room
- Radioactive waste and decommissioning
- Plant life management and aging issues.

Two specific research activities will be presented in more detail, namely RutheniumReleases (Experiments on the behaviour of ruthenium in air ingress accidents) and CostCalculation (Cost calculations and related issues with regard to decommissioning and dismantling of nuclear research facilities).

Quality in Radiation Protection

T. Bennerstedt[1], C. G. Lindvall[2]

[1]NKS; [2] BKAB

Quality in Radiation Protection Work

Carl Göran Lindvall, BKAB
Torkel Bennerstedt, NKS

Quality Seminars

- Initiated by: NSFS
- Arranged and sponsored by: BKAB and NKS
- Two seminars, both in Malmö, Sweden: November 2001 February 2004
- Two programs – same goals

QA in Radiation Protection NSFS-NKS-BKAB

- First Malmö meeting: Only nuclear installations
- Second Malmö meeting: Included medical uses

Target Audience

Radiation protection workers and officers in

- industry
- medicare
- research
- authorities

Scope

Present and future quality aspects on radiation protection work,
taking into account:

- international requirements and recommendations
- authority requirements
- operational requirements
- ALARA

Objectives

- Update on QA developments in the radiation protection field
- Identification of intrinsic similarities in QA work at nuclear power plants and hospitals
- Exchange of experiences in QA work in practical radiation protection
- Networking

Format of the Seminars

- Two full days
- Lectures
- Group work
- Plenary discussions
- Study visit to Barsebäck (Malmö-2)
- Much appreciated social program

Examples of Lectures (1)

- ICRP recommendations and practical radiation protection work
- Definition of quality
- QA in the radiation protection process
- Hospital physics, radiation protection and QA (Malmö-2)
- Environmental and quality certification

Exampels of Lectures (2)

- Issues for the future: Developments, decommissioning, waste management, final repositories
- Safety, quality and economy
- East-West cooperation: Waste, environment, radiation protection and QA

Examples of Group Work

- Decommissioning, waste
- Measurements, monitoring
- Dose reduction
- Medicare and industry: What can we learn from each other in QA and rad.prot. work?
- Economy, radiation protection, safety, ALARA
- Organizational issues, education, training
- Exchange of experience and communication to the young generation

Nordic Participation

- Both seminars were arranged in Sweden
- Some 70 participants per seminar
- Of those, about 50 were from Sweden
- The rest from other Nordic countries
- Should the venue be changed next time to achieve a more evenly distributed participation?

Questionnaire After Malmö-2 (1)

- Background of participants:
  40 industry, 10 hospitals, 20 authorities
- 1/3 participated in both seminars
- No. of lectures vs. group work was fine
- The quality of the lectures was excellent or good
- This type of seminars should be arranged with 2 – 3 yr. intervals

Questionnaire After Malmö-2 (2)

- Good forum for discussions and exchange of experiences
- Continue to encourage insight into neighboring fields, including non-nuclear applications, e.g., off-shore industry and traffic safety
- The social activities the first evening were excellent

The Future

Given the right conditions, BKAB and NKS are willing to continue the series of seminars on ”Quality in Radiation Protection Work”.

That, however, requires an external initiative, e.g., from NSFS and its coming Norwegian Board of Governors

www.nks.org

Here you will find

- documents from the two Malmö meetings
- additional information, e.g., the results of the questionnaire (Malmö-2)
- detailed information on NKS and its activities
- a link to subscribe to NKS NewsFlashes and NewsLetters

NKS-B: Introduction

S. E. Pálsson[1], T. Bennerstedt[2]

[1]Geislavarnir ríkisins; [2]NKS

NKS-B: Introduction

Sigurður Emil Pálsson,
Geislavarnir ríkisins
Torkel Bennerstedt, NKS

Objectives of NKS-B

- Strengthen Nordic radiological emergency preparedness
- Maintain and enhance competence in radioecology
- Contribute to improved communication and information tools and systems
- Create and maintain informal networks

Overall Framework

The program structure is based on three major fields:

- Research, investigations, exercises etc.
- Seminars
- Education

Creating New Knowledge

Activities aimed at generating new knowledge and competence are carried out in three fields:

- Emergency Preparedness (general as well as support tools & systems oriented)
- Measurement strategy, technology and quality assurance
- Radioecological studies

Dissemination of Knowledge and Information

- Seminars or workshops where results are presented and discussed, future activities are planned, and networks are created and expanded
- MSc and PhD educational and training programs in, e.g., radioecology and environmental measurements
- Scientific papers, technical reports, web pages, newsletters etc.

Criteria for NKS-B Activities

All proposed activities are evaluated to see if they, e.g.,

- Fit into the NKS-B framework
- Contribute to increased knowledge and networking
- Benefit financiers and end users
- Have important technical, scientific and pedagogical merits

Nordic co-operation in radioecology and radiological/nuclear emergency preparedness: Results of the NKS-B programme 2002-2005

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Abstract: This is a short overview of the NKS-B programme for the period 2002 up until this year, and some of the experience from this period of NKS-projects. Although the European Union includes most of the participating countries in the Nordic co-operation, the NKS enables a focus on Nordic needs and conditions in radioecology and emergency preparedness. The NKS work in this field is important, through maintaining competence and encouraging co-operation through contacts and networks. Experts outside the Nordic countries have also been invited to participate in certain types of NKS-B activities, such as seminar, courses and laboratory intercomparisons. Especially co-operation with the Baltic States is a fruitful addition to the Nordic NKS-work. Some of the challenges the B-programme tries to address, is the link between highly competent radiological and radio-ecological communities and the implementation into emergency preparedness and decision systems.

Introduction

The NKS programme has been going on for decades, and has justified itself at several occasions. The Chernobyl accident in 1986 still stands out as an accident that required contacts and cooperation across country borders, and where NKS networks were invaluable. At the end of 2001 the NKS board started discussing the future structure of NKS, and for this last period a number of project-leaders were substituted with two programme chiefs. There is now one R-part and one B-part, the B-part being emergency preparedness, including radioecology and emergency preparedness related information and communication needs.

The programme shall strengthen the radiological preparedness in the Nordic countries, both practical and organizational. In general, the activities can be divided into three main headings:

- Emergency Preparedness (general preparedness and specific tools)
- Measurement Strategy, Technology and Quality Assurance
- Radioecological Studies - of relevance for emergency preparedness

Contrary to previous periods, this period has been very dynamic, with many activities going on at the same time, and most projects being of one year's duration or less. This has of course increased the programme chief's coordination task, but it has also increased the number of products out of the programme, in form of reports, seminars and co-operation between projects.

Results and discussion

To illustrate the activities, we have listed the projects within the B-programme.

Table 1. An overview of ongoing and previous activities within the NKS-B programme. The last three projects (in grey) were started this year.

Acronym	*Full name*	*Type*
UrbContSem	Urban contamination seminar	Seminar
EccoMags	Nordic-EU collaboration on design and evaluation of the Resume 2002 exercise	Project
MSG-ModMeth	Co-ordination and modernization of methods for AGS (air born gamma spectrometry) and CGS (car born gamma spectrometry) measurements of multi-nuclide contamination	Project
MSG-Course	Course in advanced methods for processing ASG and CGS data and similar sets of spectral data	Course
ASS-1	Area Specific stripping for CGS and AGS	Course
Nova Course	Additional funding of PhD course in radioecology	Course
NucVess	Impact assessments of accidents with nuclear powered vessels – analysis of release mechanisms and source term composition	Project
Knowledgebase	Nuclear threats in the vicinity of the Nordic countries – A base of knowledge	Database
RadSem	Radioecology and measurement techniques	Seminar
ASSb	Area Specific Stripping of lower energy windows for ASG and CGS NaI systems. Part 2	Course
CommTech	Communication technology and emergency preparedness	Project & seminar
Cskinetic	Human metabolism of caesium	Project
Rein	Regional differences in reindeer radiocaesium contamination	Project
Metnet	Nordic network of meteorological services engaged in nuclear emergency preparedness	Project/Network
NorCMass	Nordic collaboration on the use of mass-spectrometers for the analysis of radioisotopes	Project
EMARAD	Emergency management & radiation monitoring in nuclear and radiological accidents	Project & seminar
INDOFERN	New indicator organisms for environmental radio-	Project

	activity	
ECODOSES	Improving radiological assessment of doses to man from terrestrial ecosystems	Project
Labinco	Intercomparison of laboratory analyses of radionuclides in environmental samples	Project
UrbHand	Decision Support Handbook for remediation of contaminated inhabited areas	Project
Irades	Internal Radiation Doses in Emergency Situations	Project
RadChem	Radiochemical analysis in emergency and routine situations	Project
NordRisk	Nuclear risk from atmospheric dispersion in Northern Europe	Project
SAMPSTRAT	Sampling strategy and sample preparation in emergency situations	Seminar
FOREST	Guidance for sampling in forests for radionuclide analysis and update of the Nordic forest radioecology network	Project

As the table demonstrates, the number of project is high, and there is a combination of projects, seminar and courses. In addition, many of the projects present and contribute to the common seminars that are being held, e.g. the NKS-B Summary Seminar to be held in Estonia in October this year.

Conclusions

The NKS wants to be perceived as a competent organization, both within the Nordic countries as well as in the international community. Therefore information about NKS work is important, through web-sites, technical reports, scientific papers, seminars and through international co-operation. Both the NKS homepage www.nks.org and the B-programme page www.gr.is/nks-b are recommended if you want more information about ongoing activities or project reports.

The NKS has proven to be useful for developing research areas and emergency preparedness needs that can later be brought on to larger projects in e.g. the EU. Still, the Nordic countries have specific needs or special conditions, e.g. within radioecology, that will not be addressed in EU-projects. We also know that the NKS enhances close links between the different projects and possible end-users, and will continue to focus on areas where more knowledge is needed. This justifies future NKS-work.

Why, when and how to collect samples for forest radioecological studies – an introduction to the project FOREST

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Abstract: The aims of the project FOREST are to compile a sampling guide for radionuclide analysis of northern forests and to activate a network of Nordic radioecologists. The sampling instructions for soil and vegetation will be prepared in collaboration with radioecologists and forest researchers. The compilation of guidance under the project FOREST will be implemented in future field studies and is motivated by achievements through improved reliability and comparability of new data that will be produced in those future studies. During the work with preparation of the sampling guide, key persons working with forest research will be contacted and invited to the network.

Introduction

After several well-managed studies on forest radioecology in Europe during the last few decades, the need for new data on forests is still obvious. The post-Chernobyl changes in radionuclide distribution in forests have not been measured for a sufficient variety of forest types to fulfil the requirements of environmental and dose assessments and modelling. Besides the need for representative data, there is a requirement for a representative analysis of forests of different vegetation types and structures. Furthermore, this has to be balanced against the very laborious sampling and analysis methods.

Concerning the new generation of scientists that enters this field the challenge is to collect relevant new information on time-dependent changes in forest ecosystems. To achieve this, knowing the basics of radioecology and nutrient cycling in forests are necessary. The sampling methodology has to be connected with the processes related to the transfer of radionuclides in forests. No projects focusing on sampling methodology in forest ecosystems have been carried out internationally as a Nordic Nuclear Safety Research (NKS) or EU-project earlier. This paper introduces the plan of a recently activated project FOREST running under NKS Programme B.

Sampling guide and network for forest radioecologists

Qualified sampling is a necessity in analysis of forest products and tree stands potentially or actually contaminated by radionuclides. The complex structure of vegetation and soil in forests and time-dependent changes in radionuclide contents in various forest compartments require special skills and experience to carry out suitable sampling of vegetation and soil. Vegetation needs to be sectioned in a logical way. First,

it is important to obtain comparable radionuclide contents and second, certain compartment structures are needed for radionuclide analysis in order to complete and improve the database for radioecological modelling. For calibration of external dose rates from the ground, representative samples showing the depth distribution of radionuclides and sufficient information on the site characteristics are necessary.

The results of the project will give a realistic frame for planning the accreditation of sampling methods used in radioecological field studies of forests. On the other hand, it can be foreseen that exemption levels for radionuclide concentrations in commodities presented in the Safety Guide by International Atomic Energy Agency may become a constraint in international trade of timber after a large-scale radionuclide contamination of forests (IAEA Safety Guide No. RS-G-1.7, 2004). In such a situation quality assured sampling carried out by certified field technicians using accredited sampling methods would become more important than today.

The sampling procedures for soil and vegetation for radionuclide analysis and assessment of radiation exposure will be prepared in collaboration with radioecologists and forest researchers from Finland, Norway and Sweden. The aim of the guidance is to improve the accuracy, comparability and representativeness of new data on forest radioactivity. Thereby the reliability of model parameters derived from the data, and the assessment of radiation doses after radiological accidents will also be improved.

A draft guide for sampling will be prepared by the end of 2005. The guide is planned to cover examples on sampling strategies, guidelines for site description, quality assurance, and practical instructions for collecting samples in the forest ecosystems. Site description (e.g. fertility, vegetation structure, tree stand characteristics, topography, hydrology) is essential to enable utilisation of results in modelling and exposure assessments. Sampling purposes will be, for example, national and international forest ecosystem monitoring based on recommended methods, and surveillance of human exposure to radiation and potential needs for exposure assessment of biota.

Random, grid or stratified sampling designs are usual for field studies. Other important factors to be considered are the size of samples (e.g. due to detection limits), the number of replications of samples, composite samples versus single samples, resources for sample preparation and analysis, storage of samples, availability of sample materials, timing and frequencies of sampling.

Sampling equipment will be described giving a reasonable basis for making choices between them. Avoiding cross contamination in field conditions and in the laboratory is essential in radioanalytical work. Quality assurance of all stages in sampling is one of the main aims of the study. Here also the number of samples and its statistical implications are crucial. However, costs of sample preparation and analysis often have an influence on the number of analyses.

A finalised Guide for sampling in forests is planned to be submitted by the end of 2006.

A nordic network will be formed during the work for the sampling procedures and a workshop is planned for 2006 or 2007. The FOREST project can provide a new starting point for intensified Nordic collaboration in forest radioecology. The working methods of the network could be exchange of information, and organisation of topical workshops as a forum for presentation of new research and latest results. Forestry experts will also be invited, as they need information for building their own preparedness for problems caused by radionuclide contamination.

Session XIII: Radioecology and artificial radionuclides in the environment

Basic considerations in radioecology - or: Radioecology, do we really need it?

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Abstract: Nordic radioecology has always held a strong position in the international community. There has been lots of resources put into this field, ever since the early era of testing of nuclear weapons in the atmosphere. Accidents, radiofobia and the geographical closeness to former Soviet Union has since then kept this line of science alive, some will claim. Especially the consequences following the Chernobyl-accident has been thoroughly studied. Surely, we must know everything there is to know by now. Still, some radioecologists still come up with new ideas and new projects that should be explored. An important task today is to communicate the knowledge gained, and implement them in emergency preparedness, decision support systems and framework for protecting the environment.

Introduction

Radioecology seems to be a never ending study, and can include everything from ecology, measurement techniques, sampling strategies, nuclear physics and chemistry, statistics, social sciences, computer technology, modelling, metabolism studies, nutrition pathways, phsycology, and also economy (so-called radioeconomy). And when the picture seems to be complete or understandable in one area, we just come up with a new ecosystem, new species or a new radionuclide and we can continue our studies through the whole chain of diciplines. In addition, radiation protection has formerly been based on the statement that if man is protected, then the environment is also protected. This is in many cases true, since man is one of the more vulnerable species, but it can not be applied everywhere. Thus, the recognition of the need of a framework for protecting the environment has resulted in the need for more knowledge on radiation effects on biota and the introduction of reference organisms. The new focus on climate changes also raises questions about what will happen with concentrations of radioactivity, e.g. radioactivity trapped today in the tundra, or radon exhalation in northern areas.

Radioecology is in many ways a child of the cold war. Global fallout from the nuclear weapons tests in the atmosphere in the 1950s and 1960s required attention. The first priority was to monitor the fallout, but quickly also monitoring foodstuffs to make sure that the concentrations did not exceed “maximum permissible limits” became important. It was also shown that typical Nordic conditions were vulnerable to radioactive contamination. High precipitation, unfavourable climate and agricultural conditions all contributed to high radioactivity concentrations in important foodstuffs. Especially the food-chain lichen-reindeer-reindeer herders would be exposed to radioactive fallout. During the 1960s, country-wide systems for monitoring radioactivity were established. Although increased concentrations of radiostrontium, radiocesium and radioiodine were monitored in milk, other food stuffs and even in humans, the experts did not agree on the risk and

possible effects connected to this radioactive contamination, and were afraid that e.g. mothers might stop giving milk to their children, or in other ways "behave irrational". Consequently, no countermeasures were implemented and there was no official information strategy.

As concentrations of radioactivity decreased in the 1970s, most monitoring-programmes were terminated. It was by then well established that milk and reindeer would have high concentrations of radioactivity following radioactive fallout, ecological half-lives were calculated, and uptake via direct deposition was compared to root uptake. Many radioecologists retired, or went on to other professions.

Then, one could say that the Chernobyl-accident revitalised radioecology. The accident was an accident Nordic authorities were not well prepared for. Although there had been earlier nuclear accidents, and even if they were severe accident, their long-range consequences were small. Accidents such as the Windscale accident in 1957 and the Three Mile Island accident in 1976 did not lead to widespread contamination to such an extend as the Chernobyl-accident did. Very little information on what had really happened was available in the early phase. The Nordic authorities had to deal with the situation, monitoring systems had to be set up, and practical countermeasures implemented. Although radioecology was not an important element in the early phase, it soon became evident that the problems would be of a long-term character.

Apart from the near-areas in Ukraine, Belarus and Russia, also areas in Nordic countries received substantial amounts of radioactive fallout. The radiocesium, ^{134}Cs and ^{137}Cs were the long-term problem, deposition ranging from 2 to 500 kBq/m^2. The deposition was very inhomogenous, and it was difficult to make a comprehensive assessment of the situation. After a first phase of reassuring people there would be no acute danger, it became clear that the radioactive fallout would affect both people and food production in large areas.

Some examples on concentrations that were measured the first summer following the Chernobyl-accident: Radiocesium in sheep, 40,000 Bq/kg, mushroom 45,000 Bq/kg, maximum up to 1-2 MBq/kg, reindeer 150,000 Bq/kg, and the average in reindeer grazing in the Snåsa area (middle Norway) was 70,000 Bq/kg. As the slaughtering season was getting closer, the situation became increasingly worse. The intervention limits implied that 85% of all reindeer production (545,000 kg of meat) was not fit for human consumption, and had to be destructed. In addition, 320,000 sheep had to go through special feeding before slaughter.

A whole range of countermeasures were investigated and implemented, focusing on reducing the uptake from plants to animals and reduction of intake to humans. Methods for monitoring live animals had to be developed. By using the animal's biological half-life, domestic animals could be given clean feed or special feeding and within 3 weeks the concentration in sheep would be reduced to 50% of the initial concentration. For reindeer there is an even shorter half-life of 14 days. Additives and monitoring techniques had to be tested, and feedback on how the methods were working was an important part of the work.

In order to have effective countermeasures, good knowledge in radioecology is not enough. Practical conditions at the farms, communicating with affected farmers and discussing results are also important. Only by joint discussions and exchange of experiences, one can arrive at practical solutions that work.

Countermeasures are still necessary, nearly 20 years after the Chernobyl-accident. Cow and goat milk is still above intervention limits of 370 Bq/l in some areas, if Preussian blue is not administered, and goat milk is still above limits for brown cheese production (50 Bq/kg) in many areas. Special feeding of sheep is also still necessary on a large scale. In addition, plenty of mushrooms in the grazing areas causes large variations in radioactivity in milk and meat, and makes it difficult to predict ”half-lives” because of the large variations from year to year. Radioecology studies can contribute to improving countermeasures and predict when and where measures are needed.

The countermeasures are often challenged: Why spend a lot of money on sheep and reindeer? Does the ALARA-principle apply in this case, and do we really know enough about low radiation doses and associated risks?

Today both Norway and Sweden have national intervention limits for selected foodstuffs (e.g. reindeer meat, game and freshwater fish). These special limits have also raised concern; The reindeer herders were immediately concerned: Were they not equal to others? Would anyone buy their products? Also, the limits are different in Sweden and Norway, how can this be?

The second year after the Chernobyl-accident, the average ^{137}Cs whole body concentration in male reindeer herders in Middle Norway was around 600 Bq/kg. As a group, they exceeded the maximum recommended dose limit of 1 mSv/year. Today, in 2005, the average in this group is 150 Bq/kg, still much higher than in the average Norwegian citizen, but well below the limit.

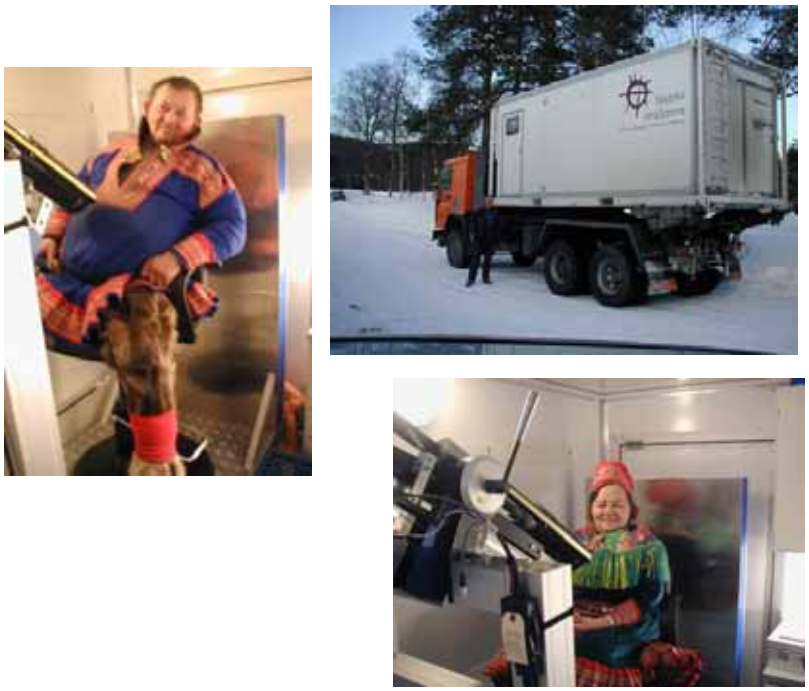

Figure 1. *Whole body measurements on selected population groups, using mobile monitoring equipment. Finland, Sweden and Norway have valuable time-series of radiocesium in man.*

The Chernobyl accident happened almost 19 years ago, and society is continuously developing. We know that local involvement, information, communication and open discussions will be very important in managing a possible future accident. In 1986, one fifth of

the Norwegian population (22 %) had little or no belief in the information given. Radiofobia is also an expression that was often used following the Chernobyl accident.

Studies on risk communication and risk perception show that in general, people tend to underestimate risk. But this optimistic bias is most common when the risk is seen as controllable. Radiation and cancer is most often perceived as uncontrollable. Many believe that authorities, and even your physician minimises risk of cancer so that you will not worry. To improve risk communication following a nuclear accident, personalising the message, use of graphical display and direct communication can enhance the comprehension and acceptance of how the situation is managed. Authorities should also accept that some people will make their own decisions on how to cope with the situation. However, if the basic problem is public mistrust in the authorities, then changing the message will not help. It takes many years to build trust. Active management of an accident of similar scale as the Chernobyl-accident will only be successful if local authorities or other persons with trust and acceptance among the population are involved and can give advice on how to manage the situation.

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The sinking of the Russian nuclear submarine K-159 and the subsequent actions of the Norwegian authorities

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Abstract: The decommissioned non-defuelled submarine K-159 was being towed to the Nerpa shipyard for dismantlement when it sank on the 30th August 2003 three nautical miles northwest of the island Kildin, less than 130 km from the Norwegian border, where it currently rests on the sea floor at a depth of 238 meters. Nine crewmembers died in the accident. Based upon information available at the time and experiences with similar accidents, the Norwegian authorities found that there was no immediate danger of cross-border contamination.

Introduction

The purpose of this paper is to sum up the course of events during and following the sinking of the submarine K-159, the Norwegian authorities immediate reaction to the accident and the subsequent actions taken by the authorities.

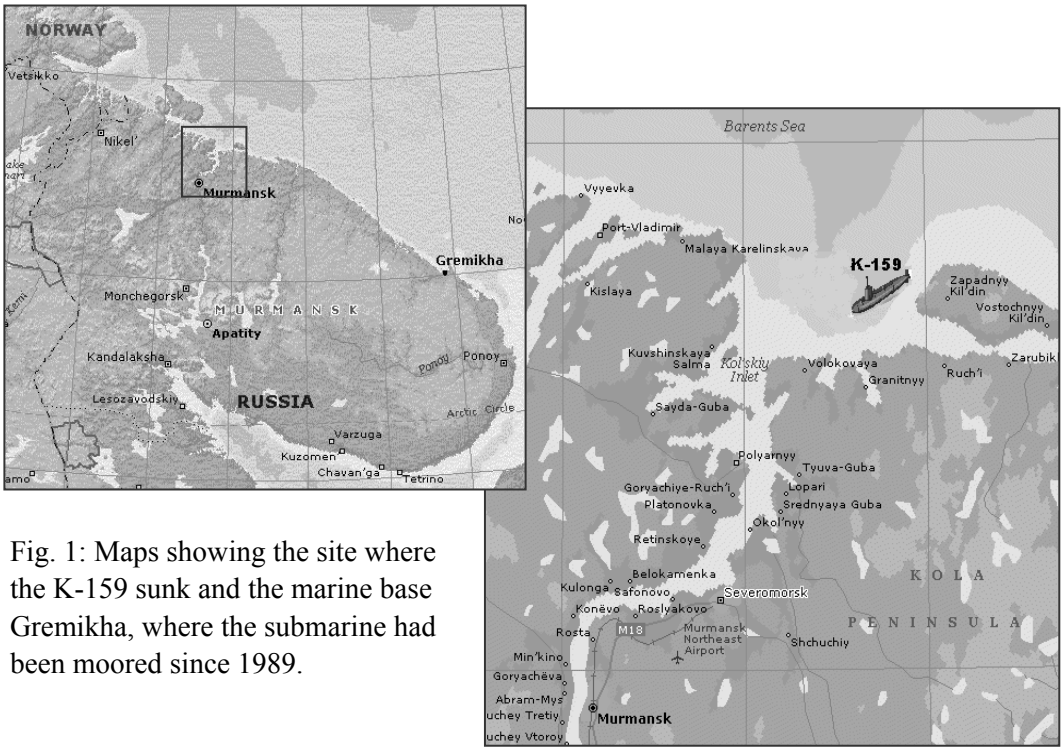

Fig. 1: Maps showing the site where the K-159 sunk and the marine base Gremikha, where the submarine had been moored since 1989.

The accident and subsequent actions

Fig. 2: K-159 leaving Gremikha. (Photo: The Russian Northern Fleet)

At 04 a.m. (local Russian time) of the 30th August 2003, the decommissioned non-defuelled submarine K-159 sank three nautical miles northwest of the island Kildin at the mouth of the Murmansk fjord. The submarine was being towed with the help of four pontoons from the submarine base Iokanga in Gremikha to the shipyard Nerpa, where it was to be dismantled. A storm was blowing in the area at the time of the accident. Several reasons have been given for the sinking of the K-159, among them that the pontoons keeping submarine afloat were torn off by the storm. Other sources claim that the main cause of the accident was leaks in several of the submarines compartments. Ten crewmembers were on board the submarine as it sunk, one crewmember was rescued and three were later found dead, the remaining crewmembers are missing, presumed still to be inside the submarine. The K-159 was a member of the Soviet Union's first generation of submarines, the Project 627 (NATO classification November), launched in 1963 and taken out of service in 1989. At that time the reactors were shut down and secured, but without the fuel being removed. Since then the submarine has been moored at the Iokanga base.

The Norwegian Radiation Protection Authority (NRPA) learned about the accident at 07 a.m. the same morning through Norwegian radio, which had picked up the news from Russian media. The emergency response organization within the NRPA immediately convened and the Crisis Committee for Nuclear Accidents was alerted to accident.

Contact was made with the Norwegian Defence, the Norwegian embassy in Moscow and the consulate-general in Murmansk to attain information about the sinking of the K-159, the condition of the submarine and its nuclear reactors. The Russian Chargé d'Affaires in Norway informed the Ministry of Foreign Affairs about accident the same morning. Furthermore the Chargé d'Affaires confirmed that the K-159s nuclear reactors were shut down, but not defuelled, and that the submarine was not carrying nuclear weapons. According to the Russian authorities no increase in radiation levels in the sea surrounding the sunken submarine could be detected.

The Crisis Committee for Nuclear Accidents

The Crisis Committee for Nuclear Accidents heads the National Preparedness Organization for Nuclear Accidents. The Crisis Committee will in the acute phase of a nuclear accident determine the actions to be taken; in the subsequent phase, it will act as advisor to the authorities. The Norwegian Radiation Protection Authority's (NRPA) director is the leader of the Crisis Committee and NRPA holds the secretariat functions. In a crisis NRPA will assist the Crisis Committee with scientific expertise, gathering information, situation assessment and collation of environmental data.

Based upon the information available at the time and previous experiences with similar loss-of-ship accidents, including the accidents with the Kursk (Amundsen 2001) and the Komsomolets, the Norwegian authorities assessed that there was no immediate danger of cross-border contamination. A rough estimate of the radionuclide inventory was made, based upon technical data available on comparable reactors. In a long-term perspective, as the barriers confining the radioactive fuel corrode and disintegrate, radionuclides will be released to the environment. Extensive studies of the radiological impact from other sunken nuclear submarines and dumped reactors show that, although the reactors poses a serious threat to the environment, there are no large-scale release of radioactive substances. The radioactive contamination for the most part is localized and no far-reaching environmental effects are found (Høybråten 2003). In accordance to this NRPA concluded that any radioactive contamination would be detected through the existing Norwegian marine monitoring programs, however fish and water sampling in the Barents Sea was done earlier than scheduled. No elevated levels of radioactivity were detected.

The NRPA sent out four emergency response announcements regarding the accident to the members of the Crisis Committee, the appurtenant board of scientific advisers, the prime minister, the ministries and the county governors. There are two elevated levels of preparedness within the National Preparedness Organization for Nuclear Accidents, namely 'Information preparedness' and 'Elevated Nuclear Emergency preparedness'. Based on a total evaluation of the situation, none of the two elevated preparedness levels were declared. Two press releases were dispatched and media inquiries responded to.

At an extraordinary meeting of the Crisis Committee the on 3rd September the NRPA informed about the accident and the actions taken in response to it. The Crisis Committee was satisfied with the assessments made by the NRPA and concluded that there was no need to take further actions at this time.

The sinking of the K-159 did not fall under the existing bilateral early notification agreement for nuclear accidents between Norway and Russia. This agreement, which is based on the IAEA Convention, concerns accidents where the radioactive release has the potential to spread to the other country's territory and result in radiation doses to the population. Thus the threshold for alerting neighbouring countries and the international community is relatively high. Norway has therefore called for the early warning threshold to be lowered. A number of events, most important the K-159 accident, have prompted this request. A Norwegian-Russian protocol concerning strengthened collaboration on emergency preparedness and early notification was signed 21. October 2003.

The condition of the submarine K-159

The submarine is 107 m long, equipped with two pressurized water reactors and the submerged displacement is 3,986 tons. Each reactor had a thermal effect of 70 megawatts. The submarine is of double hull construction with 9 watertight compartments (Pavlov 1997). The submarine had been moored at Gremikha the last 14 years before it was towed of to be dismantled. Due to problems with buoyancy towing with the support of four pontoons were the standard method of transportation to the shipyard for dismantlement for older submarines like the K-159.

The K-159 is equipped with the Russian navy's oldest model of pressurized water reactors, VM-A. At present there are about 800 kg of used fuel in the reactors, most likely uranium oxide fuel with 5-20 % enrichment. The reactors were last refueled in 1972, so

there is reason to believe that the fuel has a high degree of burn-out. This reduces the danger of a criticality accident, but also means that there are large amounts of fission products in the reactors, especially cesium-137 and strontium-90, which are highly bio-available (Ølgaard 2001).

Russian authorities informed that the total activity in the reactors were less than 7.4 PBq An accurate calculation of the radionuclide inventory is complicated by the large number of unknown factors, like the total operating time of the reactors, never the less an estimate can be made by comparing the K-159 to similar submarines and dumped reactors. Calculations made by the NRPA at the time of the accident estimated the total activity to be between 3 and 13 PBq, depending on total operating time, which are in the same range as the activity given by the Russian authorities. It is also relevant to compare the K-159 to the K-8 submarine, sunken in the Bay of Biscay in 1970; IAEA has calculated the total activity in its reactors to 9.25 PBq (IAEA-TECDOC-1242).

Raising of the K-159

Russian authorities have maintained that the K-159 will be raised, but the operation has been postponed a number of times. Last time no new time frame was given. Although much smaller than the Kursk the K-159 lies much deeper and thus it is more technically challenging to raise. The poor condition of the submarine is adding to the risk of the operating, making it even more difficult.

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Radionuclide uptake and transfer coefficients in Finnmark reindeers

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Abstract: The reindeer herding community is in general considered to be the most vulnerable population in Norway in regard to deposition of airborne radionuclides following a potential foreign nuclear accident. A more comprehensive understanding of radionuclide uptake and transfer coefficients in reindeers is important in an emergency preparedness perspective, and is currently being approached by studying differences in radiocesium contents between different reindeer herds and forage availability.

Introduction

Atmospheric tests of nuclear weapons were made by several nations during the period 1945 - 1962 and individual tests have occurred later, the latest in October 1980 (UNSCEAR, 1993). These tests resulted in local and regional deposition of fission products, non-fissioned material and activation products. Additionally, high yield nuclear detonations caused injection of radioactive debris to the Stratosphere, resulting in world-wide fallout of radionuclides, continuing even several years after the detonations.

With their close proximity to the Saami region, the Russian tests sites at Novaya Zemlya are of particular concern. The Novaya Zemlya archipelago has been an important location for Russian atmospheric tests from 1954 to 1964, and subsequently underground until 1991. In addition to stratospheric fallout, these tests may have caused local and regional fallout in the Saami Region. The content of radionuclides in the semi-natural ecosystem in most part of the Saami areas are mainly from fallout from the atmospheric nuclear weapon tests in the fifties and the sixties.

Furthermore, the accident at the Chernobyl nuclear power plant on 28 April 1986 caused considerable fallout over areas both inside and outside of the former USSR. The weather conditions were such that considerable quantities of the released radioactive material were brought by prevailing winds to different parts of the Saami region.

Of the radionuclides that were deposited, strontium and cesium isotopes were in particular considered to be of concern for human health. Their long physical half-lives of approximately 30 years and their relative similarity to natural body constituents result in relatively high uptake in animals and humans.

The Arctic nature is one of the world's most vulnerable environments with respect to contamination with radioactive fallout. The same characteristics of Arctic plants that enable them to survive in the Arctic environment, lead to an efficient accumulation and concentration of radionuclides in these species. Additionally, because of the poor nutritional stage in the semi-natural ecosystem, the contamination will stay in the upper layer of the soil and be available for the plants for decades.

The use of semi-natural ecosystems for household makes the Saami both economically and cultural vulnerable for radioactive contamination. Agriculture, forestry, reindeer herding, with supplement of hunting and fishing are still economically important. Berries are important for many families as well.

The food chain lichen-reindeer-man is important due to radioactive contamination. Reindeers in the Arctic still contain radiocesium from the deposition of radioactive fallout originating in the atmospheric nuclear weapon tests in the 60's and 70's. Reindeers are grazing lichen predominantly during winter, with a lower intake during the rest of the year. Lichen have a high ability to absorb radionuclides directly from precipitation, they grow slowly and have a long lifespan, resulting in relatively long biological half-life of the absorbed radioactivity (Tuominen & Jakkola 1973). Based on data from Westerlund et al. (1987), the ecological half-life in reindeers on lichen pastures in these areas is about eight years. The food chain lichen-reindeer-man is an important pathway of radiocesium to humans in the Arctic, particularly to reindeer herders and their families.

Material and Methods

Muscle samples from domestic reindeer (*Rangifer tarandus tarandus)* were collected from five different reindeer herding districts in Finnmark. The reindeer herding districts are located north of 69° N and between 20° – 30° E. The samples are from adult reindeers and calves, and are collected at the slaughtering season in January 2004 and 2005. Lichens (*Cladonia spp.* and *Cetraria spp*.) and soil samples are collected from the winter grazing pasture for the respective reindeer herds in 2004 and 2005. All samples are dried and counted on a high purity germanium detector.

Result and Conclusions

Radiocesium from the atmospheric nuclear weapons testing still exists in some foodstuffs in the semi-natural ecosystem in the Saami region. Radiocesium has been of considerably significance as a contaminant for people who have been using food from the nature during the last decades. The fallout was deposed relatively evenly in the sub-Arctic and Arctic regions, and of most important was the long-lived cesium-137 and strontium-90. The Cs-137 concentration in reindeer meat in 1965 was about 3000 Bq/kg (w.w.) (Westerlund et al. 1987).

The Chernobyl accident in April 1986 led to the deposition of radioactive fallout over Fennoskandia in highly variable concentrations depending upon prevailing winds and rains. About 20 percent of the Cs-137 content in reindeer from Finnmark in 1986 and 1987 had its origin from the in Chernobyl accident.

The result from our investigation shows Cs-137 content ranging from 50 – 250 Bq/kg (w.w.) in reindeer samples were the district with the lowest content had an average about 70 Bq/kg and the highest district an average about 200 Bq/kg (w.w). There were no differences between adult animals and calves.

Soil samples from the grazing areas ranges from 50 – 500 Bq/kg (d.w.) and lichen from 20 – 200 Bq/kg (d.w.).

The differences in the radiocesium content in the between the different reindeer herds may reflect differences in forage availability or the radiocesium content of the lichen. Long-term series of data from some of the areas in Finnmark shows a slight decrease in Cs-137 content in reindeers.

Lichens, which are the main dominant forage for reindeers during the winter seasons and the main source for the radioactive contamination in reindeers, had a lower radiocesium content than expected.

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Estimations of the Cs-137 distribution in a boreal forest using a traditional sampling method compared with the use of a portable NaI-detector connected to a geographical information system

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Abstract

Field-portable detectors have been used frequently in routine monitoring and in hazard assessment studies. However, there have been few attempts to carry out a careful evaluation of their potential as an alternative to sample collection and subsequent laboratory analysis, the subject of this study, conducted at the Nyänget catchment in the Svartberget Research Forest, 60 km northwest of Umeå, northern Sweden. The objectives were i) to determine the ^{137}Cs activity in recharge areas, discharge areas, shrub mire and open mire, ii) to map the geographical distribution of ^{137}Cs by means of the portable NaI detector connected to a GIS system and iii) to identify ^{137}Cs anomalies in the catchment. The study used two different methods, the portable NaI detector (GDM-40) and soil samples measured by gamma-ray spectrometry at the laboratory. The agreement in ^{137}Cs activity between soil samples and GDM-40 data is good. More than 50% of the Cs^{137} activity in all areas was found in the upper 5cm of the soil. The GDM-40 survey of ^{137}Cs at the catchment shows a trend in higher activity in the west to east direction as well as lower activity on the mire. The lower values in the south eastern part are explained by the low activity on the road. Our soil sample data indicates that the caesium lost from this catchment moves from the recharge area to the shrub mire and from the shrub mire to the open mire and then out in the stream. The topography and the climate in this catchment may be a key to those redistribution processes. Previous research in this catchment showed that the radioactive caesium in this catchment was leached from the mire to the stream and constituted a secondary source for contamination of lakes. Nylén and Grip (1997) estimated that the yearly loss of caesium from the shrub mire was 1.8% and 30% from the open mire. They were, however, not able to demonstrate any loss from the recharge areas during the period 1986 to 1989. Our results demonstrate that the GDM-40 has very good potential for making ^{137}Cs inventories and to detect ^{137}Cs anomalies within large areas. This method is much faster than a traditional soil sampling method. Soil samples can, however, not be excluded totally because they are needed for calibration of the GDM-40. The drawback of a portable detector method is that if smaller differences in activities are to be studied it may be difficult to correct for depth distribution, attenuation in soil, and contribution from activity in trees.

Radiocaesium in reindeer pasture

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Abstract: This work is part of a larger study with the aim to find out to what extent different parts of the vegetation may influence on the transfer of radiocaesium to reindeer. We have used data on ^{137}Cs in plants and lichens collected from northern Finland and central and northern Sweden. Levels of ^{137}Cs were 5-50 times higher in samples collected in northern Sweden compared to the same species collected within the Finnish reindeer herding area. Relatively higher levels of ^{137}Cs, compared to other plants, were found in plants on wetland and in lichens. The decline of ^{137}Cs over time varied between area and species. In some species no significant decline could be observed. Relatively short half-lives, 3-6 years, were observed in mosses and lichens.

Introduction

Reindeer have a relative high uptake of radionuclides due their winter diet that usually contains a high proportion of lichens, which effectively absorb radonuclides from air and precipitation (Thomas *et al*. 1992). A considerable seasonal variation and a relatively short long-term decline in the body of reindeer have been demonstrated after the Chernobyl accident (Eikelmann et al. 1990; Åhman & Åhman 1994). The decline becomes slower with time and varies between different areas (Åhman et al. 2001). The relative portion of pre-Chernobyl caesium seems to be one factor that decides the rate of the decline. To be able to understand the dynamics of radiocaesium in reindeer it is necessary to have sufficient information about radiocaesium in the vegetation that forms the diet of reindeer. In this work we have used data on radiocesium activity concentrations in some lichens and plants collected during the years 1986 to 2004 from different parts of Sweden and Finland. The work is part of a larger study with the aim to find out to what extent different parts of the vegetation may influence on the transfer of radiocaesium to reindeer, depending on the source and age of deposition, geographical location, biotope and season.

Results and discussion

The activity concentrations of ^{137}Cs were 5-50 times higher in samples collected in northern Sweden compared to the same species collected within the Finnish reindeer herding area. This can be explained by higher fallout from the Chernobyl accident in the Swedish areas. In the area of Kuusamo, Finland, the activity concentrations of ^{137}Cs recorded in plants from 1988 until 2003 were in most cases below 200 Bq kg^{-1}. There was a significant decline of ^{137}Cs in eight of the totally 13 observed species (Table 1). The decline was most rapid in moss and arboreal lichens, with effective half-lives from 4 to 6 years. In these species about 80% of the ^{137}Cs were estimated (based on their ^{134}Cs content) to be

from the Chernobyl fallout whilst in the other species only 30-50% came from the Chernobyl fallout.

Table 1. Effective half-life of ^{137}Cs and calculated average level of ^{137}Cs in 1986 in different plant species collected in Kuusamo, northern Finland

	Species *	First year of observation	Decline until 2003 $T_{1/2}$ (years)	Calculated ^{137}Cs in 1986 (Bq/kg)
Leaves:	*Betula pubescens*	1990	not sign.	35
	Betula nana	1988	13,5	54
	Salix lapponum	1988	10,7	59
Shrub:	*Vaccinium myrtillus*	1988	9,5	110
Grass:	*Deschampsia flexuosa*	1990	not sign.	33
Sedges:	*Carex spp.*	1988	9,4	390
Herb:	*Menyantes trifoliata*	1990	not sign.	240
Horsetail:	*Equisetum palustre*	1992	not sign.	53
	Equisetum fluviatile	1990	not sign.	200
Moss:	*Pleurozium scherberi*	1990	4,8	160
	Hylocomium splendens	1990	4,4	60
Arboreal lichen:	*Alectoria spp.*	1991	4,0	100
	Bryoria spp.	1990	6,0	180

The ^{137}Cs activity concentrations in plants collected in 1997 and 1998 from different reindeer herding districts (paliskunta) in Northern Finland differed significantly between species (Table 2). Highest activity concentrations were found in plants growing on wetland (*Menyantes trifoliata*, *Rubus chamaemorus* and *Equisetum*). There was no significant effect of geographical site (reindeer herding district) in this material.

Table 2. Caesium-137 in different plant species collected during the summers of 1997 and 1998 within the Finnish reindeer herding area (each sample is pooled from several sub-samples and represent one reindeer herding district and year)

	Species	No of sampels	Mean	±Std Dev	
Leaves:	*Betula pubescens*	66	38	±49	a
	Salix spp.	49	54	±38	ab
Shrubs:	*Betula nana*	72	34	±23	a
	Vaccinium myrtillus	73	70	±28	b
	Vaccinium uliginosum	76	76	±94	b
Herbs:	*Epilobium angustifolium*	41	28	±21	a
	Menyantes trifoliata	52	173	±80	c
	Rubus chamaemorus	58	201	±105	d
Grass:	*Deschampsia flexuosa*	67	26	±11	a
Cottongrass:	*Eriophorum spp.*	55	120	±90	e
Horsetail:	*Equisetum spp.*	56	234	±146	f

Mean values not connected by same letter are significantly different

The ground deposition in the area of Vindeln in northern Sweden varied between 9 000 and 18 000 Bq m^2. The highest activity concentration of ^{137}Cs (48 700 Bq kg^{-1}) was found in *Calluna vulgaris* collected in 1986. The average aggregated transfer factor (Tag) for *Calluna vulgaris* was also significantly higher than for all other plants except *Nymphaea alba* (Table 3). The plants from Vindeln were collected primarily to investigate the diet of moose. In this material, *Vaccinium myrtillus* (with an intermediated Tag), *Epilobium angustifolium* (low Tag) and *Equisetum* (high Tag) are the species most relevant in the reindeer diet. There was a significant decline of ^{137}Cs from 1986 to 1988, but not thereafter, in *Vaccinium myrtillus* collected from this area (Fig. 1).

Table 3. Activity concentration and aggregated transfer factors (Tag) for ^{137}Cs in different plant species collected in 1986 in a forest area at Vindeln, Northern Sweden

	Species	No of samples	^{137}Cs (Bq kg^{-1}) mean	±SD	Tag 1986 (m^2 kg^{-1})
Trees, twigs:	*Betula pendula*	10	227	±172	0,010[a]
	Betula pubescens	149	381	±326	0,017[a]
	Pinus sylvestris	13	551	±436	0,024[a]
Shrubs:	*Calluna vulgaris*	10	19 928	±10 740	0,886[b]
	Vaccinium myrtillus	112	1 664	±920	0,072[a]
Herbs:	*Epilobium angustifolium*	25	194	±117	0,008[a]
	Nymphaea alba	39	15 487	±11 843	0,673[bc]
Horsetail:	*Equisetum* spp.	29	13 093	±6 931	0,569[c]

Values not connected by same letter are significantly different

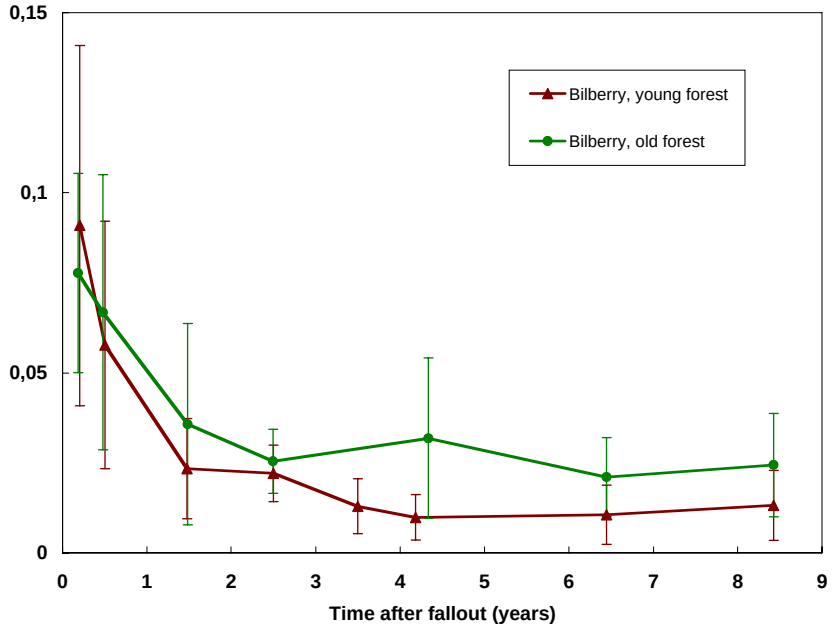

Figure 1. Caesium-137 relative to the average ground deposition (Tag, m^2 kg^{-1}) in bilberry (twigs and leaves) from an old and a young forest in Vindeln in Northern Sweden (bars represent standard deviation).

The effective half-life of ^{137}Cs in reindeer lichens (*Cladina* spp.) collected in central Sweden was shorter (3.1 years, Fig. 2) than those observed in vascular plants shown above. The rate was similar to that observed in reindeer in Sweden during the first six years after the Chernobyl accident (Åhman & Åhman 1994). The rate of decline did not change with time although the period of observation was rather long (18 years), whereas in reindeer the decline has been shown to become slower with time (Åhman et al. 2001). The relatively rapid decline in lichens, that provide most of the intake of radiocaesium to reindeer during the first years after fallout, and the slow decline in plants that have a relatively low initial uptake of radiocaesium may explain the gradually longer half-life observed in reindeer after the Chernobyl accident.

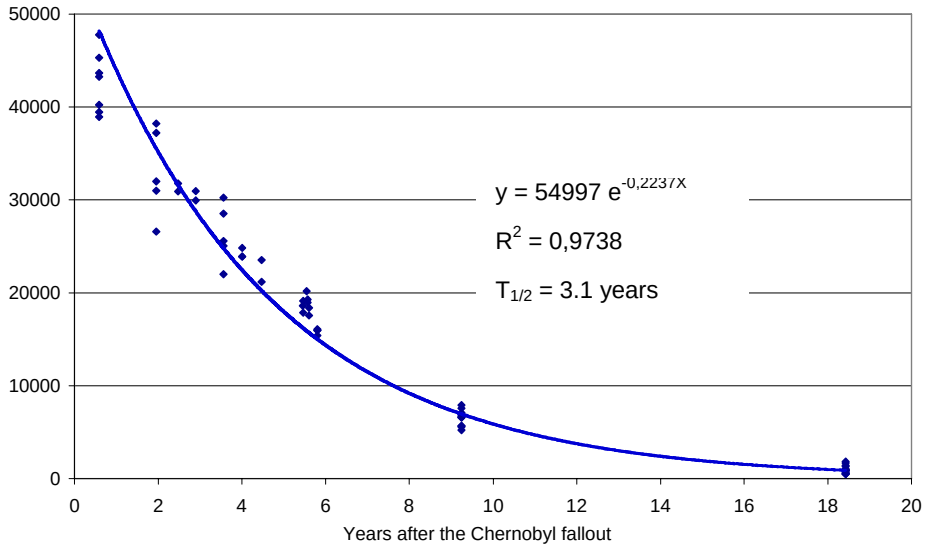

Figure 2. Caesium-137 (Bq per kg dry matter) in reindeer lichens (mainly *Cladina stellaris*) collected from a dry pine forest heath northwest of Uppsala, central Sweden.

More data will be added to this work and the results will be compared to previously published results on radiocaesium in reindeer pasture. This will help to better understand the dynamics of radiocaesium transfer from fallout to reindeer.

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GIS supported calculations of ^{137}Cs deposition in Sweden based on precipitation data

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Abstract: ^{137}Cs deposition maps were made using Kriging interpolation in a Geographical Information System (GIS). Quarterly values of ^{137}Cs deposition density per unit precipitation (Bq m^{-2} mm^{-1}) at three reference sites and quarterly precipitation at 62 weather stations distributed over Sweden were used in the calculations of Nuclear Weapons Fallout (NWF). The deposition density of ^{137}Cs, resulting from the Chernobyl accident, was calculated for western Sweden using precipitation data from 46 stations. The lowest levels of NWF ^{137}Cs deposition density were noted in the northeastern and eastern Sweden and the highest levels in the western parts of Sweden. The Chernobyl ^{137}Cs deposition density is highest along the coast in the selected area and the lowest in the southeastern part and along the middle. The sum of the calculated deposition density from NWF and Chernobyl in western Sweden was compared to accumulated activities in soil samples at 27 locations. Comparisons between the predicted values of this study show a good agreement with measured values.

Introduction

Nuclear weapons fallout (NWF) over Sweden from has occurred since the 1940s over Sweden, where most of this deposition can be attributed to tests carried out between 1962 and 1966 on Novaya Zemlya, although Chinese tests also made a contribution in the ´70s. The accumulated deposition density over Sweden from nuclear weapons tests is about 3 kBq/m^2: De Geer et al. (1978), and, in addition to this, the Chernobyl accident contributed with a deposition density varying from close to zero to well over 100 kBq/m^2, within a restricted area, SGAB (1986).

Several studies have shown that the variations in deposition density are closely related to the precipitation, e.g. Bergan, 2002; Schuller et al., 2004. The deposition of radioactive elements could therefore be estimated by relating the measured activity concentration in the precipitation (Bq L^{-1} = Bq m^{-2} mm^{-1}) at a reference site to the amount of precipitation in a geographical region (e.g. Isaksson et al., 2000; Wright et al., 1999). The point-estimates could then be interpolated to a deposition map using a Geographical information system (GIS). The development of reliable methods for predicting the spatial variation and amount of ^{137}Cs deposition following nuclear accidents or nuclear weapons explosions are of interest in e.g. emergency preparedness and by including weather forecasts from meteorological data sources a prediction of the effects of a release of radioactive elements to the atmosphere could be made. The aim of this study was to determine the deposition of ^{137}Cs due to fallout from nuclear weapons tests (NWF) over the whole area of Sweden and from the Chernobyl accident in a predefined area in the western part, visualize the results in deposition maps and compare the maps with measurements.

Material and methods

Nuclear Weapons Fallout: For the estimation of NWF three reference sites in Sweden (Kiruna, Grindsjön and Göteborg) were initially chosen since they represent areas with different mean quarterly precipitation rates and also since they had an almost complete data record of quarterly ^{137}Cs deposition density for the time period studied, 1962-1966. The period 1962-1966 was chosen since the major part of the radioactive debris was injected into the atmosphere during the periods 1952-1958 and 1961-1962 (UNSCEAR, 2000), and because no data was available prior to 1962. The deposition density per unit precipitation (Bq m^{-2} mm^{-1}) at the reference sites was found by dividing the quarterly deposition density by the quarterly precipitation. The quarterly precipitation during the years 1962-1966 at 61 weather stations, distributed over Sweden was provided by the Swedish Institute of Meteorology and Hydrology (SMHI), as well as the coordinates of the weather stations. Each reference site was then assumed to represent an area of Sweden with approximately the same quarterly mean precipitation as the reference site itself. Due to seasonal variations in the precipitation pattern these reference sites were used in different combinations to find the best way of accurately representing the deposition density over the Swedish territory. The quarterly ^{137}Cs deposition density at a weather station was found by multiplying the amount of precipitation with the deposition density per unit precipitation at the corresponding reference site for each quarter during the period 1962-1966. This was then used to calculate the integrated and cumulative deposition density for the whole period. Deposition maps were created with ordinary Kriging interpolation in the GIS software ArcView (ESRI, Environmental Systems Research Institute, Redlands, California.

The Chernobyl fallout: This study has been concentrated to an area with a radius of approximately 120 kilometres with Göteborg in the centre. Earlier, 27 reference sites have been established for environmental monitoring, which gives the opportunity to compare the results of fallout estimations to measurements performed at those sites. In a similar way as for the NWF the ^{137}Cs deposition in the area was related to the precipitation. Göteborg was chosen to be the reference site because measurements of activity concentration in precipitation and deposited material were made at the time of the accident: Mattsson and Vesanen, (1988). The main part of the total ^{137}Cs deposition came in the rain that fell in Göteborg on the 8th of May, 1986. The activity concentration in the rain water decreased rapidly with the first millimetres and at a slower rate when the amount of rain increased, which can be well approximated by a double exponential function, Equation (1). After integration the total amount of deposited cesium is a function of the local precipitation according to

$$A = 2690 - 609e^{-x_{max}/0.34} - 2080e^{-x_{max}/5.07} \quad [\mathrm{Bq/m^2}] \qquad (1)$$

x_{max} is the total amount of rain. The parameters were obtained by fitting the double exponential to the data by Mattsson and Vesanen (1988). We assume that the main part of the Chernobyl fallout was wet deposited in the rainfall on the 8th of May and that equation 1 is valid in the whole region. A precipitation map for the 8th of May were derived by ordinary Kriging interpolation in a GIS with data provided by SMHI from 46 raingauge stations relatively well spread out in the region. Equation 1 was then applied to the precipitation layer in the GIS resulting in a deposition map of the area. The deposition layer could easily be merged with the ones from the NWF calculations thus representing the total

integrated or cumulative ^{137}Cs deposition in western Sweden. The Chernobyl deposition map was compared with aerial measurements, SGAB (1986) and the total deposition, corrected for decay until 2003, with the accumulated activity in soil samples from the 27 sample locations.

Results and discussion

NWF: The deposition density of ^{137}Cs for each quarter was interpolated and summed to an integrated and a cumulative (decay corrected to 1994 and 1985, respectively) deposition density 1962-1966. The integrated deposition density can be seen in Figure 1b, where the mean is 1.853 kBq/m^2 (range: 1.416-2.695 kBq/m^2). In general, the lowest values of integrated ^{137}Cs deposition density are found in the northeast and east of Sweden where the northern area is an area with low precipitation. The highest values are found in the mountain areas in northern Sweden and in western Sweden. A comparison with the measured deposition each quarter at three sites, not used as reference sites, show good agreement.

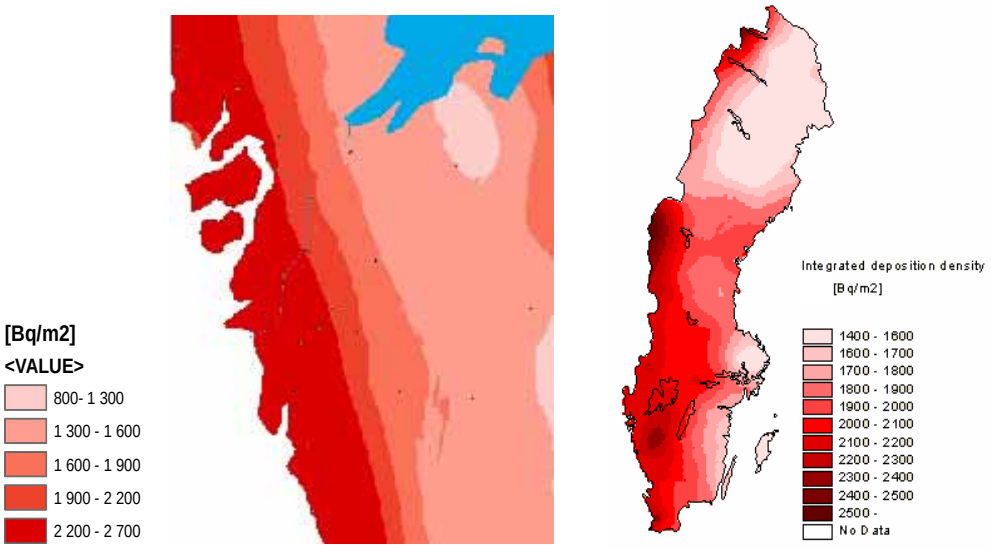

Figure 1a: Chernobyl fallout

1b: Integrated deposition density of ^{137}Cs due to NWF

The Chernobyl fallout: The mean value of the predicted deposition density over the integrated area of western Sweden is found to be 1.756 kBq/m^2 (range: 822-2.613 kBq/m^2). The highest values are found in the western parts along the coast and the lowest in an area between the coast and the lake Vänern, which can be seen in Figure 1a. The depositions from the two different sources are similar in the selected region. Calculated deposition from the Chernobyl accident was compared to aerial measurements: SGAB, 1986 and the mean ratio between calculated and measured deposition densities was 1.05±0.32. Excluding one site from the calculations because of large variations of neighbouring cells the comparison with aerial measurements yields a ratio of 0.996±0.13.

NWF+Chernobyl: The total deposition was compared to ^{137}Cs activities in soil samples and shows good agreement. The mean ratio between calculated and measured deposition densities is 1.28±0.62. If eight sites for soil sampling with large deviations are excluded the mean ratio of total calculated deposition and total accumulated activities measured in

soil samples is 1.01±0.13. The predicted deposition from the precipitation are often overestimated compared to the soil samples. The soil was sampled down to a depth of about 15 cm for most of the sites, but at some only to 12 cm. At some places this might not be enough to cover the whole inventory. There can also be locally deviations in the precipitation pattern and much could have happen to the soil in the long time period since the deposition.

The method used represents a simplified model of the deposition. It would be interesting to study the influence of other parameters such as snow, humidity, topography etc. The grouping of the weather stations might also be different if natural barriers such as mountains and ridges also are considered. With knowledge of the dependence of the activity concentration in rain on the amount of precipitation and activity concentration in air it would be possible to predict the deposition based on measurements of the activity concentration in air and meterological forecasts.

Ackowledgement

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Plutonium in the environment at Thule, Greenland, from sampling in 2003

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Abstract: The plutonium contamination of the Thule environment in Greenland resulting from an airplane crash in 1968 was investigated from samples collected in August 2003. Sediment sampling was based on information from acoustic sonar investigations of the seabed. Additional marine samples covered seawater, seaweed and benthic animals. Terrestrial samples were taken of soil profiles near Narssarssuk some 25 km south-west of the Thule airbase. Analyses of plutonium focus on the occurrence of hot particles and their importance for evaluating the total inventory of plutonium in the marine sediments.

Introduction

On January 21, 1968, an American B52 bomber carrying four nuclear weapons crashed on the sea ice 15 km west of the American air base at Pituffik, Thule Air Base. Both the aircraft and the weapons disintegrated on impact. The wreckage and the remains of the weapons were cleaned up after the accident. About 6 kg plutonium was dispersed at the accident according to American information. The plutonium distributed on the sea ice, about 3 kg, was removed. After the clean up operation and the melting of the ice the following summer, finely dispersed plutonium remained in the marine environment, primarily on the seabed under the point of impact. Since 1968 several sampling campaigns have been carried out resulting in estimates of the seabed inventory of plutonium (^{239}Pu and ^{240}Pu) of about 1.4 TBq corresponding to about 0.5 kg (Dahlgaard 2001). However, a re-evaluation focusing on hot particles in the sediments indicates that the inventory could be significantly underestimated (Eriksson, 2002).

The present investigation of the plutonium contamination at Thule is based on a sampling campaign carried out in August 2003. The scope of this investigation includes an estimate of the inventory of plutonium in marine sediments with special focus on radioactive particles, a mapping of bottom sediments in the most contaminated area based on acoustic sonar, an investigation of the transfer of plutonium to marine biota, and an investigation of plutonium in the terrestrial environment. Bylot Sound and surrounding areas are shown in Fig. 1. This work is supported by the Danish Environmental Protection Agency.

Sampling

The sampling was carried out during 14-27 August 2005 from the research vessel FS Adolf Jensen. The expedition was headed by research specialist Henning Dahlgaard; additional participants were laboratory technician Svend K. Olsen from Risø National Laboratory, scientist Mats Eriksson from Institute for Transuranium Elements in Karlsruhe, senior geologist Jørn Bo Jensen and senior engineer Peter Trøst Jørgensen both from the Geological Survey of Denmark and Greenland (GEUS), and the crew of Adolf Jensen headed by skipper Flemming Heinrich.

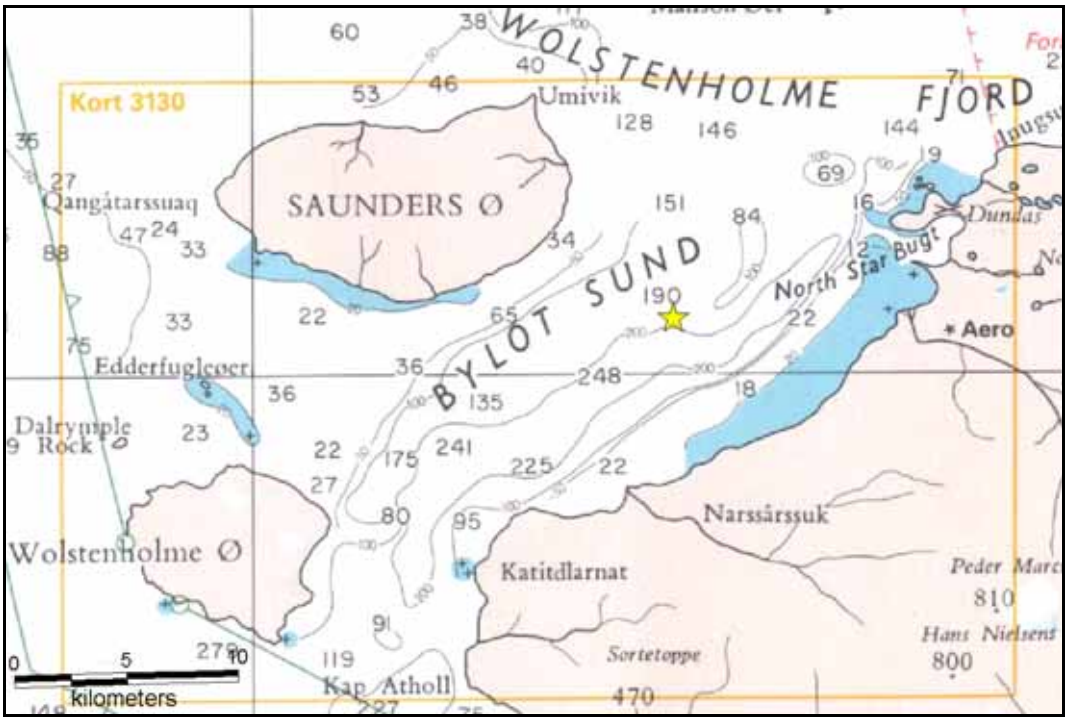

Fig. 1. Map of Bylot Sound showing the site of the aircraft accident (star).

Sediment mapping: The sediment mapping was carried out by GEUS from FS Adolf Jensen and included recording of bathymetri and characterisation of the seabed. Bathymetri, sediment profiles and surface seabed mosaic were mapped by chirp and side-scan sonar. Classification of sediments included identification of soft and hard bottoms and surface irregularities due to large objects and iceberg scour. The bathymetri of Bylot Sound is characterised by a north-south going ridge with water depths less than 100 m and a southern channel partitioned in a northern and southern basin with water depths of up to 250 m. The seismic grid used for the sediment mapping is shown in Fig. 2.

Sediments: Sediment samples were collected from 31 locations. Ten of these locations are stations from which sediment samples were obtained successfully at previous campaigns. The 21 new locations were selected based on the sediment mapping carried out by GEUS. The high success rate of this sampling demonstrates that the information obtained from the sediment mapping was extremely valuable considering the considerable problems from previous sampling campaigns to obtain useful sediment cores due to hard bottom conditions.

Biota: Benthic biota samples were collected by pulling a Sigsbee trawl along four 500 m transects. Three transects were in the contaminated area in Bylot Sound at 200-250 m depth, and the fourth transect north of Saunders Island in Wolstenholme Fjord at 135 m depth. Samples from a seal shot in Bylot Sound were collected of flesh and liver.

Seawater: Seawater samples were collected from the water column at three depths: 5 m under the surface, and 35 m and 15 m above the seabed. The samples were collected from four locations: one outside the contaminated area south of Wolstenholme Island serving as a background reference, one from north of Saunders Island as a low-level station, and two from near the accident site. The samples were filtered (Millipore 0.45 µm). Expectations were to find accident-related plutonium in the deep water near the accident site, but not in surface water or at the two other locations.

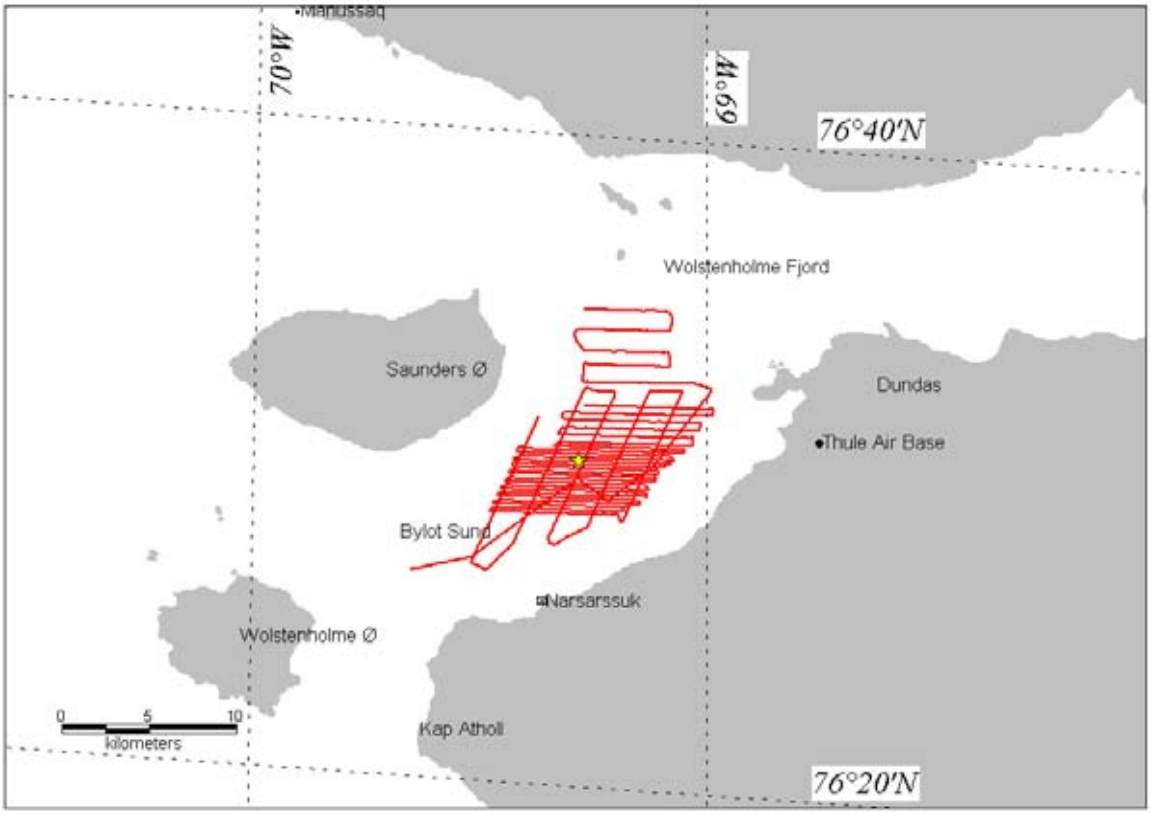

Fig 2. Seismic grid for the sediment mapping in the central part of Bylot Sound.

Seaweed: Twelve samples of seaweed were collected from coastlines at Narssarssuk south of the accident site, at Dundas Mountain, at the north coast of Wolstenholme Island and at the Eider Islands.

Soil: Soil samples were collected in the Narssarssuk area. Within small areas of one square metre 3 separate soil profiles were collected from eight locations, i.e. 24 profiles in total. Elevated levels of alpha radioactivity were found in the Narssarssuk area in spring 1968, but at that time it was not determined if this radioactivity was caused by local naturally occurring radioactivity, i.e. thorium or uranium isotopes, or due to uranium or plutonium from the aircraft accident.

Results

Seawater: Results on plutonium in seawater are given in Fig. 3, which shows concentrations of 10 mBq ^{239,240}Pu m^{-3} and below at the reference site south west of Bylot Sound. Concentrations in Bylot Sound are significantly higher in near bottom waters, up to 40 mBq m^{-3} and mainly associated with particles. However, the samples of surface water in Bylot Sound show little or no influence from accident plutonium. Accident plutonium was identified by an isotopic ratio between ^{238}Pu and ^{239,240}Pu lower than that of fallout plutonium. The location in Wolstenholme Fjord north of Saunders Island is influenced by accident plutonium from Bylot Sound with concentrations higher throughout the water column than at the reference site and showing high fractions of plutonium on particles. The seawater data illustrate that plutonium is remobilised from the seabed in Bylot Sound on particles and transported to the north out of Wolstenholme Fjord into Baffin Bay.

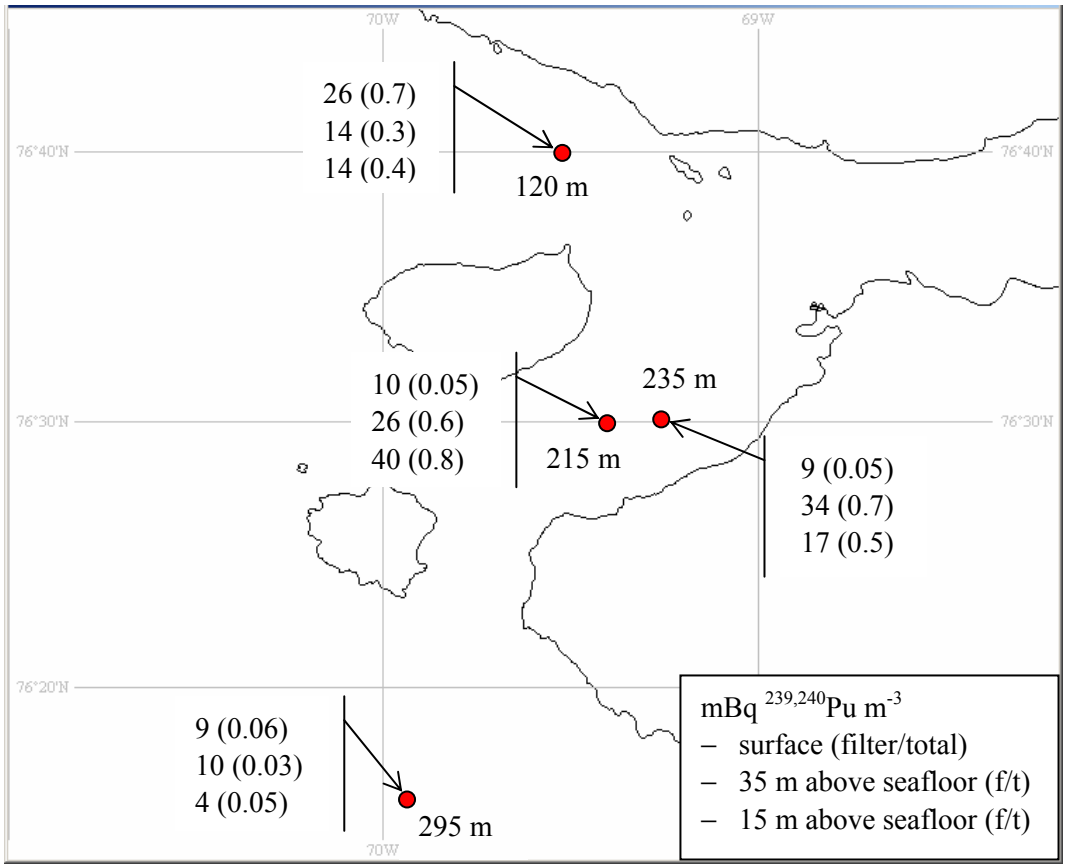

Fig.3. Plutonium in seawater at four locations sampled during 19-21 August 2003. Samples were taken at three depths for each location. Concentrations are given in mBq ^{239,240}Pu m^{-3}; ratios of Pu on particulate to total are given in brackets.

Seaweed: Plutonium concentrations in seaweed were found in the range 0.1-0.4 Bq ^{239,240}Pu kg^{-1} dw. The concentrations at Eider and Wolstenholme Islands and at Dundas were in the range 0.1-0.2 Bq kg^{-1} dw while concentrations at Narssarssuk were consistently higher, 0.2-0.4 Bq kg^{-1} dw. This shows that accident plutonium is transported to seaweed in surface waters of Bylot Sound.

Soil: Plutonium in soil at 8 locations around Narssarssuk showed depth-integrated concentrations from 20 to 900 Bq m^{-2}. The lower range of 20-50 Bq m^{-2} found at one location represents mainly fallout from atmospheric nuclear weapons tests, whereas the higher values are due to accident plutonium. Accident plutonium is inhomogeneously distributed in the soil samples and clearly associated with particles. Single particles were identified in the soil with activities up to about 30 Bq ^{239,240}Pu.

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Potassium, rubidium and caesium in fungi

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Abstract: Samples of mushrooms and soil were collected in a forest ecosystem close to Nuclear Power Plant at Forsmark, Sweden. The soil were fractionated in bulk soil, rhizosphere, soil-root interface and fungal mycelium and the concentration of K, Rb and Cs were determined. The K concentration increased from 605 mg kg^{-1} in bulk soil to 2,750 mg kg^{-1} in mycelium and 39,500 in fruitbodies of fungi. The corresponding values for Rb was 2.5 mg kg^{-1} in bulk soil and 191 mg kg^{-1} in fruitbodies of fungi. For Cs the corresponding values were 0.21 mg kg^{-1} for bulk soil and 3.9 mg kg^{-1} in fruitbodies. In fruitbodies of fungi good correlation was found between the concentration of K and Rb or of Rb and Cs, but not between K and Cs. Yoshida found similar correlation and concluded that the mechanism of Cs uptake by fungi may be different from that of K.

Introduction

K, Rb and Cs belong to the first group of the periodic table and are called alkali metals. They are all very typical metals forming ions with one negative charge. Potassium is a very abundant element ranking seventh amongst all of the elements in the Earths crust with a mean concentration of 21,000 mg kg^{-1}. Rb is also fairly abundant with about 90 mg kg^{-1}. Cs is rather rare with 3 mg kg^{-1} / Enghag 2000/. Most mineral soils except those of a sandy nature show comparable high concentrations of K – about 1.5 % by weigth. The major part of this element is held rather rigidly to the soil. In most of the soils K is not the limiting minerals for plant growth.

The hydrated ion of Rb is rather similar in size as K and are supposed to compete for the same "carrier" during uptake to plants. The mean concentration of Rb in soil has been reported to be 50 mg kg^{-1} /Coughtrey and Thorne 1983/. The Cs concentration in soils is about 5 mg kg^{-1} both with a rather large variation /Szabo 1979/. In soils with clay, Cs almost irreversible is bound to the clay minerals. The concentration of Cs seems to be rather constant in the upper soil profile, 0 – 20 cm, with slightly higher concentration in the Of and Oh layer /Yoshida et al. 2004/. Obviously the K concentration in soil will affect the plant uptake of both Rb and Cs. However, the plant uptake of these elements seems to depend on a rather complex concentration-dependent relation rather than a silmple competion.

In fruitbodies of fungi the concentrations of K, Rb and Cs are usually at least one order of magnitude higher than found in plants growing at the same site. Very high activity concentrations of ^{137}Cs have been reported by many authors but also very large variation both between different species but also within the same species. At least for Cs a rather large part of the pool of Cs in soil – 10 to 30 % of the inentory in soil- is usually found in the fungal mycelium. It has been suggested that the ability of fungi to "accumulate" radiocaesium is attributed to high influx and low efflux rates /Dighton et al 1991/

Material and Methods

The study area was located in a forest ecosystem close to the Nuclear Power Plant at Forsmark, Sweden. The main soil parent material was sandy od clayey till and the organic content in the upper part of the profile varied from 36 to 98 %. The forest types were mainly Norway spruce forest or mixed spruce and Scots pine forest. The stand was approximately 70 – 100 years old. The plant species in the field layer indicate that the soil is rather nutrient rich.

Sampling was performed during September to November 2003. We started by collecting fruitbodies of certain fungi species and then we collected the corresponding soil samples to a depth of 10 cm close to the site of fruitbodies. In the laboratory an aliquot of the soil sample (30 to 50 g) was used for preparation of fungal mycelium which was performed under microskop (64 x) using pincers and adding destilled water to the soil. The method is described in more details in Vinichuk and Johanson /2003/. After preparation the mycelium samples were dried at 35^{o} C to constant weigth.

The soil samples were also fractionated using the method described by /Gorban and Clegg 1996/. The soil was gently sieved through 2 mm mesh forming the bulk soil. The rest of the soil was gently squeezed with the fingers resulting in a further separation of more soil from the roots on the 2 mm mesh giving the rhizosphere fraction. The residue was the soil-root interface fraction.

Analysis of K. Rb and Cs were performed after digesting of the material by 5 ml HNO_3 and 0.5 ml 30 % hydrogenperoxide and analysed by ICP-AES or ICP-SFMS . These analysis were performed by Analytica, Luleå, Sweden. All results were expressed mg kg^{-1} dry weigth. The concentration ratios – mg kg^{-1} in mycelium or fruitbodies divided by same in builk soil or mycelium were also calculated

Results and discussion

As can be seen in Table 1 there at least for K and Rb a clear increasing accumulation of the nuclides when we compare the bulk soil with the fruitbodies, The presented results are median values and the variation between the studied species of fungi were quite large. The ratio of the K concentration in fruitbodies divided by the K concentrations in bulk soil found in 12 species of fungi showed a median value of 65.3 with a variation from

Table 1. Concentration (median value) of K, Rb and Cs in mg kg^{-1} in bulk soil, rizosphere, soil-root interface, mycelium and fruitbodies

	K	Rb	Cs
Bulk soil	605	2.52	0.21
Rhizosphere	806	4.3	0.32
Soil-root interface	3,280	6.47	0.2
Mycelium	2,750	12.8	0.51
Fruitbodies	39,500	191	3.9

26.6 and for Hypholoma capnoides up to 104.7 for Cantharellus tubaeformis. For Rb the variations of the CR were from 13.1 for Hypholoma hapnoides up to 675.7 for Sarcodon imbricatus. The corresponding variation for Cs was from 2.6 for Lactarius determinus and 258.8 for Sarcodon imbricatus, thus 2 order of magnitude differences. The variation increased from K to Rb and further to Cs.

Table 2. Concentration ratio mg kg^{-1} in fruitbodies (or mycelium) divided by mg k^{-1} of bulk soil (or mycelium)

	K	Rb	Cs
Mycelium/bulk soil	4.5	5.1	2.4
Fruitbodies/bulk soil	65.3	75.8	18.6
Fruitbodies/mycelium	14.4	14.9	7.6

As can be seen in Table 2, Rb showed the highest concentration ratios in all the comparisons performed and Cs the lowest concentration ratios. There seems to be some biological control of the translocation of K, Rb and Cs from the mycelium to the fruitbodies.

There was a significant correlation between the concentration of K and Rb in fruitbodies of fungi ($r = 0.51$, $P < 0.05$). There was also a significant correlation betwwen Rb and Cs in fruitbodies of fungi ($r = 0.91$, $P < 0.01$). However, there was only a veak correlation between the concentrations of K ad Cs in fruitbodies of fungi. More or less the same correlation coefficients for concentrations of K and Rb and for Rb and Cs in fruitbodies were reported by Yoshida and Marunatsu /1998/. They also reported rather weak correlation between K and Cs in fruitbodies of fungi and they concluded that the mechanisms of the Cs uptake by fungi may be diffrent from that of K. Our data seemed to confirm such statement. No correlation between K and ^{137}Cs in friutbodies of fungi were also found in sudies of 28 species from Austria /Ismael 1994/.

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Migration studies of ^{137}Cs from nuclear weapons fallout and the Chernobyl accident

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Abstract: The vertical migration of ^{137}Cs originating from nuclear weapons fallout (NWF) and the Chernobyl accident has been studied at 27 reference sites in western Sweden. An attempt to describe the present depth distribution with an alternative solution of the Convection-Dispersion Equation (CDE) with a puls-like fallout as initial condition was made. The actual depth profiles in the soil samples were fit to a sum of the CDE for both NWF and Chernobyl debris. The magnitudes of the fallouts were estimtated from precipitation calculations and GIS-mapping, leaving two free parameters (convection velocity and effective dispersion constant) for performing the fit of the depth profiles. In some cases using only two paramters is not sufficient to achieve an accurate representation for the depth profile, indicating an over- or underestimate of the magnitude of the fallout. In these cases the magnitude of the fallout is also varied. The fitted depth profiles were used to correct in situ measurements from the same locations for the actual depth distribution, showing good agreement with the accumulated activities in soil samples.

Introduction

In situ measurements are an efficient way to measure the amount of radioactive elements in the ground, but to be able to perform them in a correct way the depth distribution of the radionuclides has to be known. The assumption about the depth distribution is the largest source of uncertainty when performing in situ measurements according to ICRU (1994). The vertical migration of radionuclides is also of importance to be able to predict the potential radiological impact, e.g. if the migration is slow the radionuclide will give rise to external doses for a long time and to plant uptake by the root system and further transport in the ecosystem. The vertical migration of the antrophogenic radionuclide ^{137}Cs originating from nuclear weapons fallout (NWF) and the Chernobyl accident has been extensively investigated. Fresh fallout is often approximated by a plane surface source, but if a plane surface source is used in the calculations the activity per unit area will be underestimated due to the downward migration because the photons are attenuated in the soil. The equivalent surface deposition, Finck (1992) can be used as an approximation and can afterwards be corrected for the actual distribution by multiplying a factor. Many studies have shown that an exponential decrease of the ^{137}Cs activity concentration with depth is a good approximation the first years after the deposition, e.g. Beck (1966), Isaksson and Erlandsson (1995). However, the model is often no longer accurate after longer time periods and the profile tends to get a maximum at another depth than at the surface and often looks like the ones in Figure 1. The transport mechanisms in the vertical migration of ^{137}Cs are often assumed to be convection and diffusion and the depth profiles are therefore often explained by different solutions of the Convection-Dispersion Equation (CDE), e.g. Bosssew and Kirchner (2004).

In this study a model based on the CDE with an alternative way to solve the equations has been used. The transport mechanisms are assumed to be convection and diffusion and sorption as interaction mechanism of ^{137}Cs in liquid and solid phase, see Bossew and

Kirchner (2004). The actual depth profiles in soil samples collected at 27 reference sites in western Sweden were fitted to the solution of CDE with respect to both Chernobyl fallout and NWF. In situ measurements performed at the same locations as the soil sampling were corrected for the actual depth distribution and the results were compared to soil samples.

Material and methods

In this study 27 reference sites have been established for environmental surveillance in western Sweden defined as a region with a radius of approximately 120 kilometres with Göteborg in the centre. At each reference site in situ measurements and soil sampling have been performed in 2003.

In situ measurements: The measurements were performed with a HPGe detector (GC3019 Canberra Semiconductors N.V, Belgium, efficiency 32.6% and resolution 1.78 keV at 1332 keV), connected to an MCA device (digiDart, Ortec Inc., USA) placed 1 m above the ground, looking downwards. The sampling time for the acquired spectra was between 900 s and 1800 s. A calibration made according to Isaksson and Vesanen (2000) was used. The activities of ^{137}Cs were given as equivalent surface activity and later corrected for the actual depth distribution.

Soil sampling and analysis: Soil sampling was performed at each reference site using a metal corer with a diameter of 8 cm down to a depth of 15 cm. At each site, three cores, placed in the corners of a triangle with 60 cm side, were taken and cut in slices at 0-2 cm, 2-4 cm, 4-6 cm, 6-9 cm, 9-12 cm and 12-15 cm depth. The soil from the same depth in each of the three cores was then combined to a general sample from that site. All layers were weighed and dried at 104 ºC for about 20 hours until the weight of the samples ceased to decrease. After homogenisation and mixing the samples were put into 60 ml plastic containers for measurements with a well-calibrated HPGe detector, (PGC3419, Detector systems GmBH, Germany, efficiency 38 % and resolution 2 keV (FWHM) at 1332 keV), surrounded by a 10 cm thick lead shield. The measurement time varied between 23 h and 72 h. A background spectrum was acquired, which was subtracted from the sample spectrum. The measured ^{137}Cs activity was decay corrected to the time of the soil sampling.

The depth profiles from the soil samples were then fitted to an alternative solution of the CDE by Bossew and Kirchner (2004) assuming that the transport mechanisms are convection and diffusion and sorption as interaction mechanism of radionuclides in liquid and solid phase. The solution to the CDE with a pulse-like fallout can be seen in equation (1).

$$C(x,t) = J_0 e^{-\lambda t}\left(\frac{1}{\sqrt{\pi D t}} e^{-(x-\upsilon t)^2/(4Dt)} - \frac{\upsilon}{2D} e^{\upsilon x/D} erfc\left(\frac{\upsilon}{2}\sqrt{\frac{t}{D}} + \frac{x}{2\sqrt{Dt}}\right)\right) \text{ [Bq/cm}^3\text{]} \quad (1)$$

where J_0 (Bq/cm^2) is the initial deposition from the Chernobyl accident or the NWF, λ (years^{-1}) is the radioactive decay constant, D (cm^2/year) and ν (cm/year) are called the apparent dispersion coefficient and convection velocity, respectively; x (cm) is the linear depth and t the migration time (years). This solution can be applied to either Chernobyl cesium or NWF, but the cesium in the soil samples from 2003 cannot be separated according to their origin. Therefore the total activity in the layers was fitted to a sum of the

two according to equation (2) divided by the mean soil dry density for the layers resulting in the activity concentration given in Bq/kg. All activities were given in dry weight.

$$C_{tot} = C_{chernobyl} + C_{NWF} \quad (2)$$

The parameters ν and D were considered to be constant in the depth interval and in time. The migration time, t, is the time between the deposition event and the soil sampling date. Here the time periods since the deposition of the Chernobyl cesium are considered to be the 8th of May 1986 in western Sweden, Mattsson and Vesanen (1988) and 1st of January 1965 for the NWF. $J_{0,Chernobyl}$ and $J_{0,NWF}$ are the depositions resulting from the two sources assumed to have been deposited in one event. Since the Chernobyl fallout was deposited almost in one single event in the rainfall on the 8th of May 1986 in the area in western Sweden, it can be approximated by a pulse-like deposition. The main part of the NWF was deposited between 1962 and 1966 so the NWF has also been approximated by a pulse-like deposition due to the a long time that has passed since the depositions. The deposition from the two sources has been determined from the precipitation in Almgren et al., 2005. The fitting was made partly with the calculated values on the deposition only varying the two parameters ν and D, and partly by varying all four parameters.

The fitted depth distributions were then used to calculate the factor relating the equivalent surface activity to the actual activity distribution according to Finck (1992) and the results were compared to the accumulated activity in soil samples at the same location.

Results and discussion

For most of the locations the fitting of the observed depth profiles with the special solution to the CDE is considered to be good. Some examples of measured and fitted depth distributions can be seen in Figure 1. For most of the locations the fit has been done by only varying ν and D using the initial fallout from the calculations based on precipitation. For those which cannot be well fitted only varying two parameters, all four parameters are chosen to be varied. This is the case for those locations where the deposition based on the precipitation calculation has been under- or overestimated and was chosen for six sites. The migration is a rather slow process with a mean value of ν= 0.11±0.073 cm/year and D=1.11±0.56 cm2/year.

The comparisons of the depth corrected in situ measurements and the accumulated activities in soil samples show a good agreement. The mean ratio of the corrected in situ activity and the accumulated activity in soil samples is 1.00±0.25. The soil has been sampled to a depth of 15 cm in most cases but sometimes it was only possible to reach a depth of 12 cm. This might not be enough to cover the whole inventory for all places which can contribute to a high ratio and much could have happen to the soil in the long time period since the depositions.

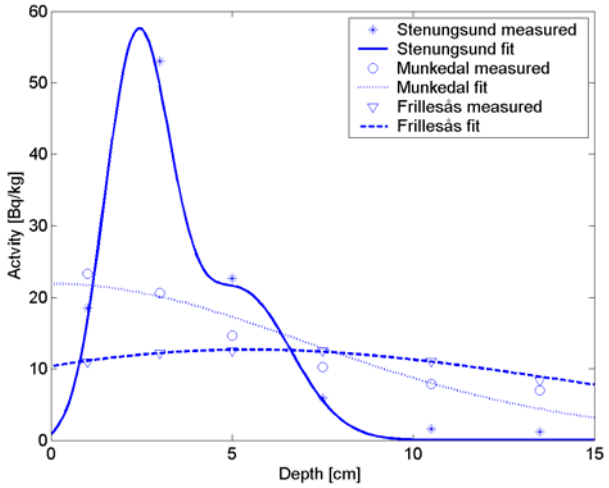

Figure 1: Depth profiles observed in soil samples fitted to a solution of the CDE.

Acknowledgement

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Cs-137 concentrations in the uppermost humus layer in the Baltic countries in 2003

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Abstract: Monitoring of anthropogenic gammanuclide concentrations in the uppermost 3 cm humus layer was carried out on samples collected in 2003 at 55 sites in Estonia, at 67 sites in Latvia and at 66 sites in Lithuania. The ^{137}Cs Bq/kg concentrations were low when a single high value of 2050 Bq/kg d.w. from north-easternmost Estonia was excluded. In all three countries the mean concentration varied between 140-160 and the highest concentrations were 400-530 Bq/kg. The calculated mean Bq/m^2 concentrations varied between 550-720, except at the NE Estonian site with a value of 12300 Bq/m^2. This high Estonian value corresponded to the Chernobyl fallout concentrations measured in 2001 in the Leningrad Region in the Barents Ecogeochemistry project. Slightly higher concentrations were located in the western parts of Latvia and Lithuania.

Introduction

The monitoring of anthropogenic radionuclide concentrations in the humus layer in the Baltic countries was carried out as a part of the Barents Ecogeochemistry Project (Salminen et all. 2004), the aim of which was to provide the authorities and other involved parties with a basis for assessing the current state of the environment. This report represents the results of gammanuclide measurements made on the 188 humus layer samples collected in 2003 in Estonia, Latvia and Lithuania.

Materials and methods

The uppermost 3 cm of the O horizon (humus layer) was sampled during 12.7. - 5.10.2003 at 55 sampling sites in Estonia, at 67 sites in Latvia and at 66 sites in Lithuania in accordance with the Field manual for the Barents Ecogeochemistry Project (Gregorauskiene et all. 2000). The samples were collected on flat areas, and consisted of a composite sample from an area of about 50 m x 50 m. The final samples consisted of a minimum of 10 cylindrical (86.5 cm^2), 3-cm thick sections of the O horizon. The humus layer represented the organic material located immediately below the living vegetation, and included the litter layer. All living vegetation was removed. The co-ordinates of the sampling sites were measured by GPS.

The humus samples were dried at ambient room temperature at the Kuopio laboratory of the Geological Survey of Finland (GSF), and sieved through a 2-mm nylon sieve to remove large roots and stones. The samples were measured in the Regional Laboratory on Northern Finland of STUK - Radiation and Nuclear Safety Authority. The samples were packed in 550 ml Marinelli measuring containers. Three small samples from Estonia had

to be packed in 100 ml 'Tamro' plastic measuring containers. All the samples were measured by low background gammaspectrometry.

Results and discussion

The ^{137}Cs Bq/kg d.w. concentrations (Table 1) were low and relatively constant in all the three Baltic countries, and varied from 23 - 530 Bq/kg. Only one exceptional high concentration of 2050 Bq/kg was measured in a sample from NE Estonia near Sillamäe (N59.4165 E28.0854). Traces of the Chernobyl accident specific ^{134}Cs isotope were detected in this sample, as was also the case in two other samples from NE Estonia, 4.0, 1.6 and 1.4 Bq/kg, respectively.

Table 1. Mean ± 1δ standard deviation, minimum and maximum ^{137}Cs concentrations (Bq/kg d.w.) in the uppermost 3 cm humus layer in Estonia, Latvia and Lithuania in 2003, (n) number of samples. Variation of measurement time.

		^{137}Cs Bq/kg d.w.				
country	(n)	mean ± 1 d	min	max	exceptional concentration	measurement time min
Estonia	55	155 ± 90*	23	400*	2050	113 - 3857
Latvia	67	140 ± 76	66	530		102 - 1051
Lithuania	66	145 ± 94	37	470		100 - 1058
* without the single exceptional high ^{137}Cs concentration						

The NE Estonia concentration is also comparable to those measured in the Barents Eco-geochemistry project (Salminen et all. 2004) in 2001. In the Leningrad Region the concentrations varied from 29 to 5300 Bq/kg. In Southern Finland the ^{137}Cs concentrations were higher, varying from 55 to 11500 Bq/kg (AMAP Assessment 2002, 2004).

The availability of sufficiently accurate data on the total mass of the humus sample and of the total area of the cylinders used in sampling made it possible to estimate the amounts of ^{137}Cs Bq per square metre in the humus layer from the Bq/kg concentrations. The calculated ^{137}Cs Bq/m^2 amounts are presented in Figure 1 in the form of a site-specific thematic map produced by MapInfo.

When the single exceptionally high value 12300 Bq/m^2 was omitted, the mean calculated value in Estonia was 550 Bq/m^2. The area most affected by the Chernobyl accident was the NW corner of the country. Extremely uneven Chernobyl deposition in Estonia has been reported by Raelo et all (1995).

In Latvia the mean calculated value was 630 Bq/m^2. Higher values were obtained for seven locations in the western part of the country, the highest value being 3400 Bq/m^2 for a sample collected near Liepaja (N56.5583 E21.1584).

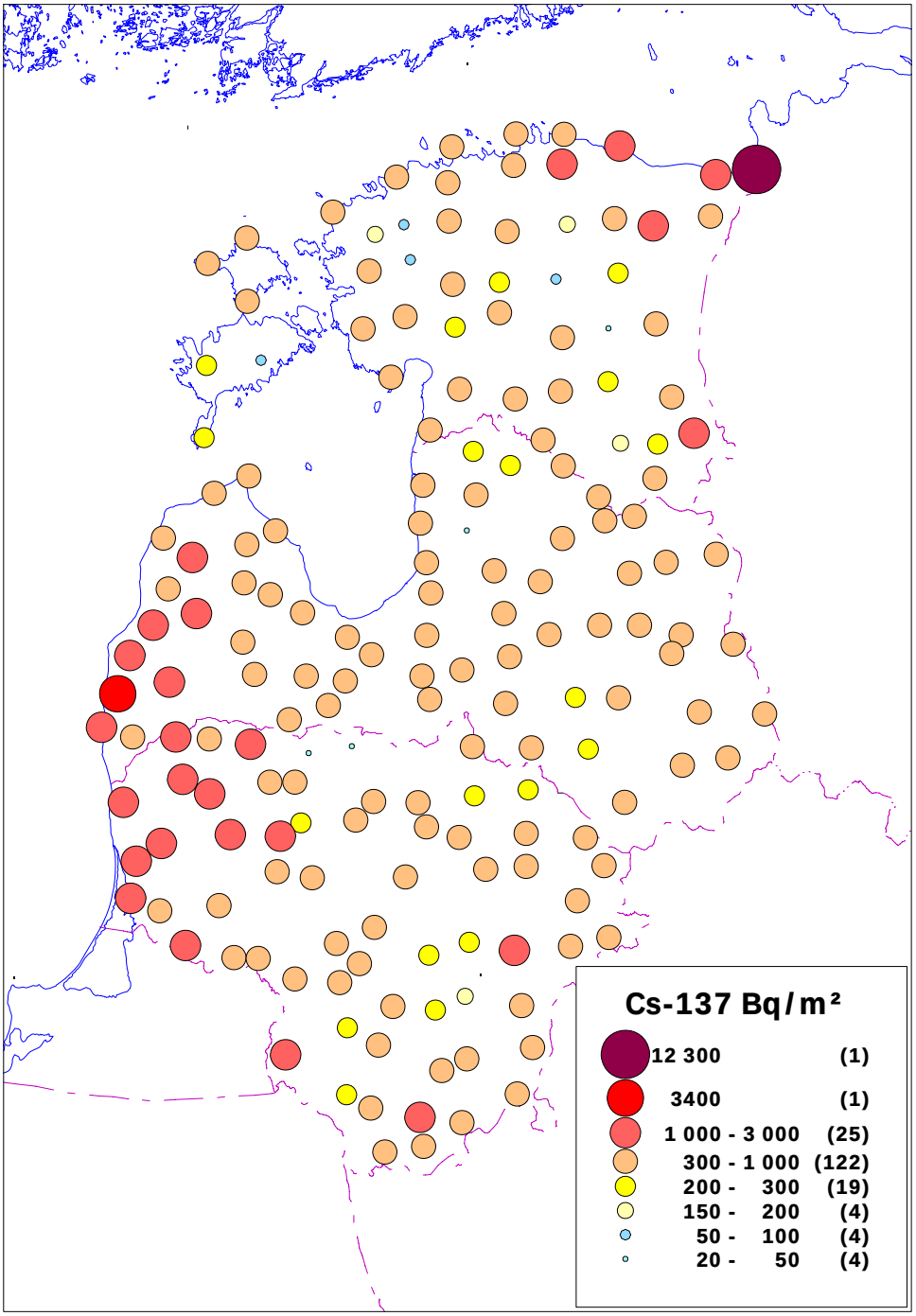

Fig. 1. ^{137}Cs values, Bq/m^2, in the surface humus layer in Estonia, Latvia and Lithuania in 2003.

In Lithuania the mean calculated value was 720 Bq/m$^{2.}$ The higher values of 1000 - 3000 Bq/m^2 were mainly located in the western part of the country, as was the case in Latvia. Traces of ^{60}Co, were however detected in a Lithuanian sample (N56.3074 E22.1347), with a ^{137}Cs concentration 200 Bq/kg. Because no Co was detected in the other part of the same sample, it may have been due to the presence of a small ^{60}Co particle.

Compared to the map in the Atlas of Caesium Deposition on Europe after the Chernobyl Accident the highest ^{137}Cs Bq/m^2 values still occurred in the north-easternmost corner of Estonia, while in Latvia and Lithuania higher values occurred only in western parts of the country.

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Comparison of different methods for determining ^{137}Cs ground deposition level

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Abstract: The aim of this work was to compare different methods for determination of ^{137}Cs ground deposition level. Soil sampling, with subsequent laboratory analysis, was performed on two sites in a boreal area, as well as *in situ* measurements using an HPGe detector. Two different efficiency calibrations were used for the detector: both a semi empirical one and one utilizing Monte Carlo simulations of the detector and source matrix. Measurements with a portable NaI(Tl) detector incorporated with a geographical information system were also made at both sites.

Introduction

Commonly efficiency calibrations for HPGe *in situ* measurements are made using more or less empirical models. At FOI a semi empirical calibration method has been developed and used for some time. However, the speed of modern computers has also facilitated calibrations using Monte Carlo simulations of the detector and source matrix. The aim of this work was to evaluate the accuracy of these two calibration methods by comparing resulting values for ground deposition level with results obtained from traditional soil sampling.

The use of Monte Carlo simulations has also been applied to the calibration of a portable, sodium iodine based detector system.

Measurements

Field tests were performed in a boreal area in the vicinity of Gävle with relatively high ^{137}Cs ground deposition levels from the Chernobyl fallout. Two measurement sites were chosen: one in alder forest and one in spruce forest.

On each measurement site 17 soil samples were collected in a cross pattern, with 2 meters between samples. Each soil sample was sectioned into three layers in order to obtain some information about the caesium depth distribution. At the lab all samples were analysed using HPGe detectors, after first being weighed, dried and homogenised.

In situ measurements were performed on both sites with an HPGe detector (36% efficiency relative a 3”x3” NaI) with 1800 seconds counting time. The ^{137}Cs ground deposition levels were calculated, both using a semi empirical calibration method, and from Monte Carlo simulations of the detector and source matrix.

The portable detector system (GDM-40) used was based on a 3”x3” sodium iodine crystal, a data logger and a GPS antenna. The efficiency calibrations for this system were also calculated via Monte Carlo simulations. Stationary measurements of about 2 minutes each were made with this system on both sites.

Results and discussion

The soil samples collected in the alder forest shows a larger variation in the ^{137}Cs content, as well as a higher total activity, than the samples from the spruce forest (Figures 1 and 2).

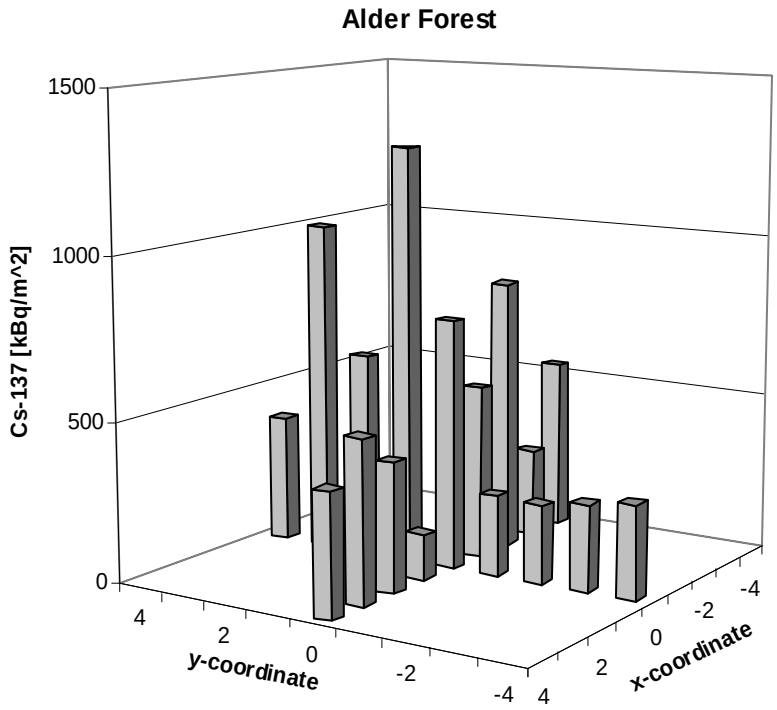

Figure 1: *^{137}Cs content in soil samples collected in the **alder forest**, recalculated to the corresponding ground deposition level in kBq/m^2.*

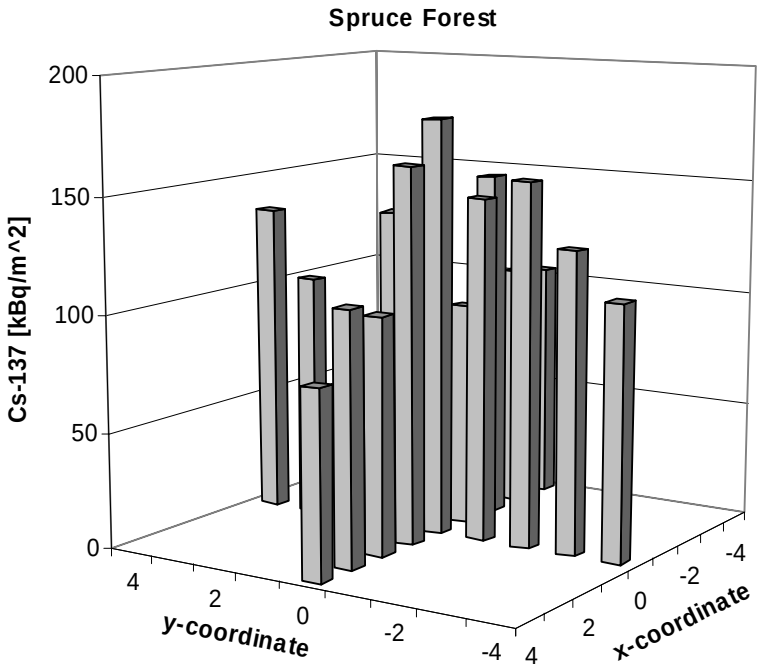

Figure 2: *^{137}Cs content in soil samples collected in the **spruce forest**, recalculated to the corresponding ground deposition level in kBq/m^2.*

The different methods used to estimate the ^{137}Cs ground deposition show very good agreement (Table 1 and Figure 3).

Notable is the ability of *in situ* measurements to average out the large variation in ground deposition levels calculated from soil samples. It is also obvious that if one is to rely solely on soil samples, a rather substantial number is required in order to achieve some statistical certainty.

Table 1. Ground deposition level values for the two sites from soil samples and *in situ* and GDM-40 measurements.

Results	Soil samples*	In Situ†	In Situ‡	GDM-40‡
Alder forest	400 (270-630)	420	430	491
Spruce forest	110 (104-148)	111	122	106

*) Median values with inter-quartile range in parentheses
†) Calibrated using a semi empirical method
‡) Calibrated using Monte Carlo simulations

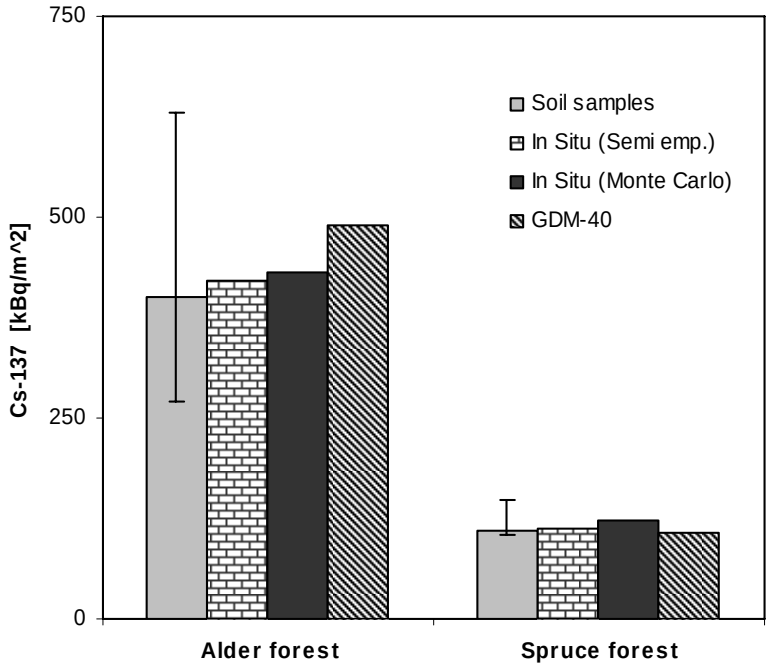

Figure 3: *Resulting values for ^{137}Cs ground deposition level in kBq/m^2 for the different measuring techniques used. For soil samples the median values are shown, with error bars representing the inter-quartile range.*

Modelling the radioecological transfer of radiocaesium to three Swedish populations

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Abstract: UNSCEAR has presented a terrestrial transfer model for predicting the annual average concentrations of radionuclides in important diet components and in humans based on the annual ground deposition from atmospheric fallout. The model is composed of three components, the principal component representing the direct contamination of fodder and crops, and the two remaining ones reflecting delayed transfer from previous fallout accumulated in soils. The model is intended to be used for protracted deposition of radionuclides and has been applied to observed time-patterns of ^{137}Cs body burdens in three different Swedish populations subjected to transfer from continuous nuclear weapons fallout during the 1960s and 70s. The results show that the best agreement with observed body burden data is obtained by applying an effective ecological half-time, $T_{eco,eff}$, for nuclear weapons fallout ^{137}Cs of 0.8 years for urban individuals in the South of Sweden, 2 years for urbans living in the Stockholm area and 4 years for reindeer herders in the middle of Sweden. When compared with the ecological half-times observed in the urban population after the Chernobyl fallout these values appear to be shorter.

Introduction

Many studies (e.g; Ågren, 1998) have been conducted on the ecological half-time of ^{137}Cs, $T_{eco,eff}$, in various Swedish populations, which are known to exhibit a varying long-term transfer of the radionuclide. In most studies $T_{eco,eff}$ has been deduced from either log-normal plots of observed in-vivo body burden vs time, or from exponential curve fits of the data. However, these methods are not altogether suitable when applied to the time patterns of body burden responding from continuous atmospheric fallout, since both short and long-term transfer add to the human intakes, and therefore the exponential slopes may appear longer than in case of a single deposition event such as the Chernobyl fallout.

UNSCEAR (UNSCEAR, 1982) has presented a model that takes into account the delayed transfer of older fallout as well as the direct transfer of fresh fallout on crops and pasture. The model has previously been used when assessing body burden data on an international level for nuclear weapons fallout. In this study the aim has been to use this transfer model on body burden data of ^{137}Cs from three different human populations in Sweden during the 1960s and 70s, in order to reconstruct the transfer components and to deduce a more representative value of the ecological half-time. The deduced values are also compared with $T_{eco,eff}$ values found in these populations after the Chernobyl fallout.

Material and methods

An empirical model for the transfer of nuclear weapons fallout through terrestrial pathways has been used by UNSCEAR; mainly aiming at predicting the annual average of ^{90}Sr and ^{137}Cs concentrations in human diet. In case of continuous fallout over many years, the cumulated deposition in soils results in a delayed aggregated transfer from ground to human diet and man, and the observed values of the body burdens are therefore

a sum of a short-term and long term components (Eq. 1). Due to the relatively short biological half-time of ^{137}Cs in humans, typically about 3 months (Leggett, 1983), one may assume that a semi-equilibrium is established between the ^{137}Cs concentrations in diet and human body tissue.

$$Y(t_n) = b_1 * A_{dep}(t_n) + b_2 * A_{dep}(t_n\text{-}1) + b_3 * \Sigma\, A_{dep}(t_0)\, e^{-\ln 2/T_{eco,eff} * (t-t0)} \quad \text{Eq. 1}$$

Y= Average ^{137}Cs concentration observed in human population during year tn

A_{dep} (t) = Annual deposition of ^{137}Cs [Bq kg-1] in year t

b_1, b_2 and b_3 = Transfer coefficients to be fitted to the regression

$T_{eco,eff}$ = ecological half-time of ^{137}Cs in aggregated components of diet due to long-term transfer including the physical decay of the radionuclide

A regression of the observed body burdens, $Y(A_{dep},t)$, vs the regional deposition of ^{137}Cs, $A_{dep}(t)$, can be used to extract the transfer coefficients b_1, b_2, b_3 and $T_{eco,eff}$ using the following three steps; Step 1: The regression function applied to observed values of Y yields the values of b_1, b_2 and b_3 giving maximum explained variance, R^2, for a given value of $T_{eco,eff}$. (Non-linear estimation), Step 2: Previous procedure is repeated for various values of $T_{eco,eff}$. The R^2 as a function of $T_{eco,eff}$ in Eq. 1 is plotted, Step 3: The transfer coefficients together with the $T_{eco,eff}$-value giving the best fit (highest R^2) is selected and applied to the observed Y-values.

Eq. 1 has been applied to the ^{137}Cs body burdens in three Swedish populations investigated in the 1960s and 70s using statistical software. Two of these populations are categorized as urban inhabitants, living in metropolitan areas in the south and middle of Sweden, whereas the third population consists of reindeer herders living in the middle of Sweden. The annual deposition $A_{dep}(t)$ has been reconstructed by combing Danish time-series of annual fallout from 1950 and onwards (Aarkrog et al., 1995), combined with a modified grid of regional pattern of annual precipitation in Sweden (SMHI, 2001).

Results and discussion

The highest correlation to observed body burden data using regression in Eq. 1 is found when applying a $T_{eco,eff}$ of 0.8 years for urban populations in south of Sweden. The corresponding values for urban inhabitants in the Stockholm area and in reindeer herders are 2 y and 4 y, respectively. The results of the scanning of the maximum R^2 vs. $T_{eco,eff}$ using the two-component transfer model are illustrated in Fig. 2. A further analysis shows that no significant difference when using only a two component transfer model for all the populations considered, thus indicating that the middle term in Eq. 1 is redundant in this particular data set. The regression data together with the observed body burden ^{137}Cs concentrations using a 2-component transfer model is given in Fig. 3. It is evident that model fails to predict body contents over a long-term probably due to a long-term ecological half-time of up to 10 y. Due to the lack of data points in time this is difficult to confirm; since adding components to Eq.1 will result in redundancy in the regression.

From Fig. 2 it is seen that for the urban populations in Sweden there appears to be no significant $T_{eco,eff}$ value that dramatically improves the regression fits, whereas for the reindeer herders an obvious increase in the R^2-value is found by setting $T_{eco,eff}$ at about 4 years. This reflects the uncertainty of the model when applied to agricultural ecosystems, where many diet components origin from products outside the area of the province. For reindeer herders the well-known transfer chain lichen-reindeer-meat-man is so dominant

that the uncertainty in the other transfer components to the reindeer herders is less dominating. In Table 1 the deduced $T_{eco,eff}$ values are compared with the corresponding values from the Chernobyl fallout.

Figure 2. Correlation coefficient (Adjusted Pearsons R^2) vs applied $T_{eco,eff}$ in UNSCEAR transfer model.

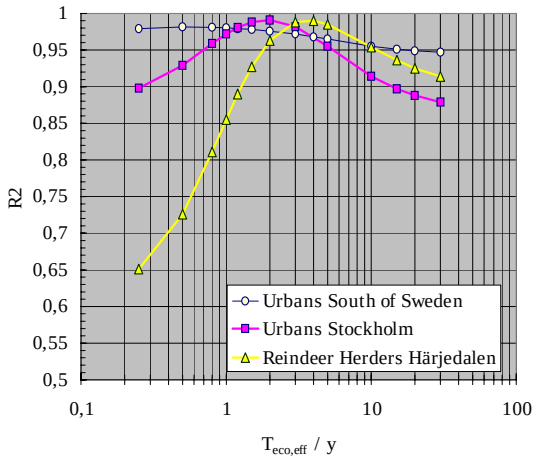
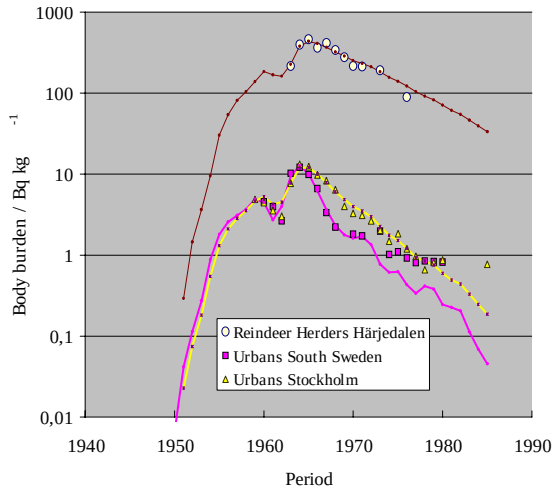


Table 1. Effective ecological half-time [y] extracted using two different methods in three Swedish populations.

Results	Nuclear Weapons fallout		Chernobyl fallout
	Using Eq. 1	[a] Curve fits – median population ^{137}Cs conc.	[a] Curve fits – median population ^{137}Cs conc.
Urbans South Sweden	0.8	1.2±0.1	2.4±0.2
Urbans Stockholm	2	3.0±1.6	2.8±0.5
Reindeer herders	4	4.8±0.3	1.4±0.1

a Values taken from the Swedish national database of human body burden compiled under the Swedish environmental goal program (Swedish Government, 2001).

Figure 3. Observed body burdens of ^{137}Cs and regression fits using $T_{eco,eff}$ values taken from fig. 2 in three Swedish human populations.



Conclusions

- Long-term transfer of ^{137}Cs to humans in Sweden can be de-composed and reconstructed from time-series of observed body burdens and average regional ground deposition values using the UNSCEAR-model.

- Values of $T_{eco,eff}$ deduced according to the empirical expression used by UNSCEAR tend to be shorter than what is obtained when using exponential curve fits to observed data.

- 3-component models appear to be redundant compared with a 2-compoenent model of transfer, due to inherent uncertainties of the deposition and transfer pathways.

- $T_{eco,eff}$ values appear to be higher among Swedish urbans after the Chernobyl fallout than in the 1960s and 70s.highest correlation.

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Forest models of the RODOS system as assessment tools for intervention after radionuclide contamination of forests

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Abstract: The forest models FDMF and LCMforest of RODOS PV6.0 are presented as assessment and support tools for the implementation of countermeasures related to forestry. A hypothetical accidental contamination of a forested area in western Finland provided a suitable case. Model results show that berry and mushroom pickers and forest workers would be subject to higher individual doses than people on average and that the highest collective doses would incur from consumption of wild berries. Criteria for access restriction and banning or processing of wild food were postulated and the interventions evaluated in terms of averted collective doses and costs. The options for increased availability of acceptable timber were assessed, namely soil improvement and timed felling.

Introduction

The forest food chain and dose model FDMF and the new countermeasure assessment model LCMforest were integrated into RODOS PV6.0 during the RODOS Migration project of the EC's 5th Framework Programme (Ehrhardt, 2004). FDMF was initially developed by STUK and IRSN during the 4th FP for assessment of individual and collective doses through ingestion of wild foods and external exposure from accidentally contaminated forests (Rantavaara et al. 2001). FDMF was subsequently improved by STUK during the 5th FP and the new model LCMforest developed to show the need for intervention in forestry and to calculate avertable doses and costs from implementation of countermeasures, as reported of a preliminary version (Rantavaara and Ammann, 2002).

This paper demonstrates the consequence assessment with FDMF and countermeasure investigation with LCMforest by using a case study. The effectiveness of banning and/or processing of wild food and access restrictions to forests were compared. Predictions to support optimal timing of tree felling operations are presented. The effect of soil improvement on the availability of acceptable timber was also assessed.

The forest food chain and dose model FDMF contains modules for calculating the initial distribution of deposited radionuclides and a compartment module for calculating the time dependent activity concentrations in the following layers of forest: crown, trunk (divided into bark and wood), understorey, top soil, soil (bioavailable), and soil (unavailable). The considered processes are weathering, foliar absorption, litterfall, root uptake, surface runoff, dissolving and fixation. Additional modules calculate kerma rates and activity concentrations in stem-wood, berries, mushrooms and game meat. Finally, individual ingestion and external doses and collective doses are estimated. For this study the database was adapted to Finnish conditions.

The forest countermeasure model LCMforest uses results of FDMF to assess the need for intervention and the effects of a countermeasure in terms of collective averted doses and costs. The need for intervention is based on criteria for the acceptability of wild foods

and timber, and access to forests. Default values are suggested for foodstuffs and external dose according to EURATOM (1987) and Ministry of Interior, Finland (2001), but the user can modify them freely. Effectiveness of certain countermeasures for reduction of contamination can also be adjusted. Model results are the suggested region and duration of intervention, avertable doses, and costs from implementation.

The contamination scenario for demonstrating the use of the models was based on a hypothetical release from the Olkiluoto nuclear power plant in western Finland. It was assumed that a severe core-damaging accident happened in mid-May, and that about 1% of the iodine and caesium inventory was released to the atmosphere. It was raining when the plume dispersed over the region. This led to the contamination of large forest areas.

Results and discussion

FDMF provides the user with an assessment of the effective lifetime doses for various population groups (Table 1). In our scenario forest workers were subject to the highest health risk from the stay in forests and pickers from their high consumption of berries. FDMF provides more insight into the actual exposure conditions. It reveals for example that although deposition of ^{131}I was about 10 times higher than that of ^{137}Cs the highest contribution to the external dose came from the caesium isotopes, and about 10 percent of the lifetime dose was accumulated within the first year and half of it within 10 years. The collective dose relates to the expected number of additional cancer deaths that would be caused by the utilisation of the contaminated forests (Table 2). This information marks the baseline when assessing the benefit from various protective or mitigating measures.

Thematic maps best convey the spatial extent of the problem. Figure 1a for example shows the spatial dose-rate distribution. LCMforest can now be used to assess the duration of intervention, i.e. the time the intervention levels were exceeded (Figure 1b). Within the 20 km radius it might take several years that dose-rates were again below the chosen intervention level 10 μSv h^{-1}. Furthermore, averted collective doses can be calculated and provide valuable estimates for defining an optimized intervention level. In our case the averted collective dose would be 10 manSv if the intervention level was 10 μSv h^{-1} and negligible for an intervention level of 100 μSv h^{-1}, which is the default value. The lower value was chosen in order to be able to better demonstrate this option.

Restriction of the consumption of wild food was more beneficial than access control. For example, when all berries exceeding the maximum permitted levels (2000 Bq kg^{-1} for isotopes of iodine and 1250 Bq kg^{-1} for isotopes of caesium and others with half-life >10d) that the EC regulation (EURATOM 1987) suggests, were banned, 590 manSv might be averted (Table 3). For mushrooms and game meat the averted doses were lower than for berries, but considerably higher than for access control. Banning consumption would avert most of the health effects that otherwise were expected (Table 2). On the other hand it incurred costs as the corresponding production was lost.

Intensified processing of foodstuffs can also be considered as an option. When wild food is normally processed it is assumed that 50% of the caesium activity remains in the prepared mushrooms, 80% in cooked berries and 90% in cooked game meat. In this case study 10%, 50% and 30% were assumed to remain in mushrooms, berries and game meat, respectively, after intensified processing. The measure can be evaluated in terms of the same attributes as used for banning, and this measure would be rather cost-effective (Table 3). LCMforest provides the option *Intensified processing or banning* for the case that

some of the wild food still exceeds maximum permitted levels. This option rejects processing where it would not reduce the concentration below permitted levels. In these areas wild food is banned.

Long-term contamination of wood by ^{137}Cs and harms thereof can with LCMforest be reduced with two options, namely soil improvement and timed felling. Soil improvement decreases the uptake of caesium by trees and thereby improves the availability of acceptable timber. For results in Figure 2 it was assumed that soil improvement was applied one year after contaminating deposition and the activity concentration was decreased by 50% within 7 years. Wood becomes slowly contaminated by root uptake (Figure 3a). This fact can be exploited by premature harvesting. LCMforest shows the minimum time needed to advance harvesting so that wood is acceptable (Figure 3b). Criterion for acceptable timber was an activity concentration of 500 Bq/kg.

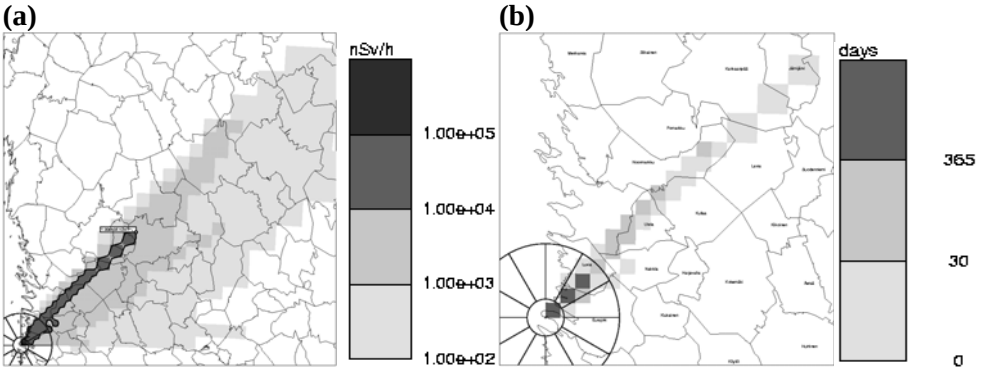

Figure 1. External dose-rate in forests. Isoline shows 10 μSv h^{-1}. *Note:* nSv h^{-1} is used here but μSv h^{-1} in the text **(a)**. Duration of external dose-rate exceeding 10 μSv h^{-1} **(b)**.

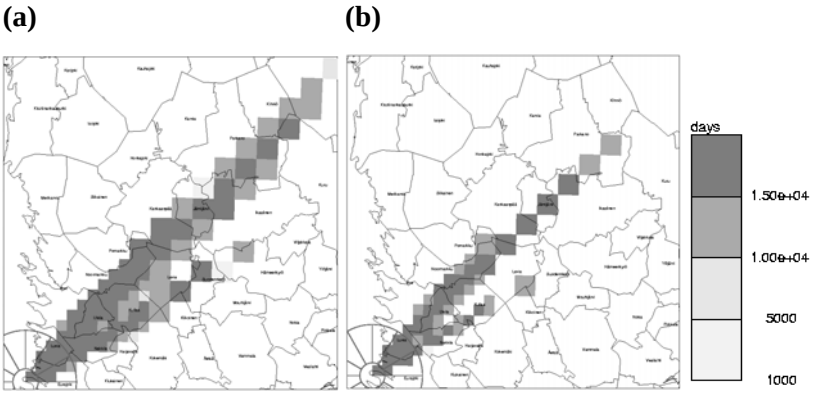

Figure 2. Duration of wood contamination by ^{137}Cs >500 Bq kg^{-1} in case that no countermeasure is carried out **(a),** and in case that soil improvement is applied **(b).**

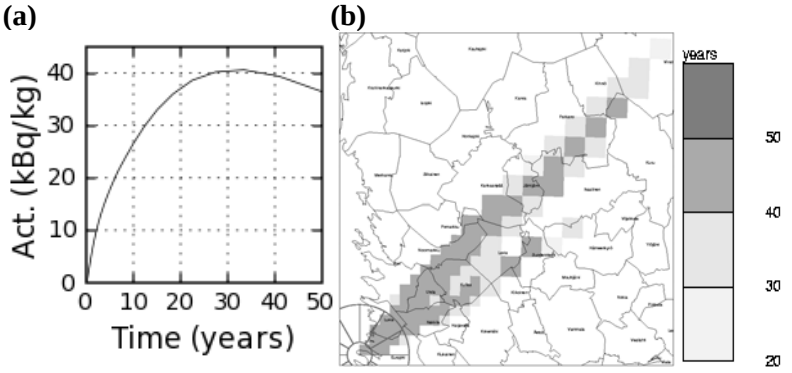

Figure 3. Maximum activity concentration of ^{137}Cs in wood (Bq kg^{-1}) as a function of time **(a)** and the minimum time to advance the harvesting so that 500 Bq/kg will not be exceeded **(b).**

Table 1. Maximum effective lifetime dose (mSv) to certain population groups by pathway.

Population	External	Berries	Mushrooms	Game meat
Adults	75	280	32	21
Collectors	83	1100	150	37
Forest workers	1200	280	32	21
Hunters	110	520	46	300

Table 2. Collective dose (manSv) for certain population groups by dose pathway.

Population	External	Berries	Mushrooms	Game meat
Adults	370	780	160	100
Collectors	41	310	74	18
Forest workers	6	1	0	0
Hunters	11	29	5	30

Table 3. Costs and benefit of countermeasures related to wild foods.

Countermeasure by type of product	Averted collective dose (manSv)	Costs (mill. euro)	Cost/dose ratio (1000 euro per manSv)
Berries			
Ban	590	26	44
Intensified processing	220	4	18
Intensified processing or banning	510	14	27
Mushrooms			
Ban	140	8	57
Intensified processing	110	1	9
Intensified processing or banning	110	1	11
Game meat			
Ban	76	10	132
Intensified processing	50	1	14
Intensified processing or ban	64	3	50

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Household methods to reduce ^{137}Cs contents of mushrooms

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Abstract: High radiocaesium contents in different species of mushrooms have been observed in areas contaminated by radiocaesium deposition after the Chernobyl accident in 1986. There has been no significant reduction in the ^{137}Cs contents of mushrooms during the past ten years, besides via radioactive decay. The internal radiation dose received via mushrooms can be reduced by processing mushrooms before consumption. Various household methods were studied to find out their efficiency to reduce ^{137}Cs contents of mushrooms. The methods tested were the same as normally used in cooking. The tests were made for the species of edible mushrooms widely consumed. The retention factors for the treatments tested were in most cases 0.2-0.3. The efficiency of treatments in reducing the ^{137}Cs contents increased with larger water volumes and prolonged treatment times.

Introduction

Since 1986, the year of the Chernobyl accident, many species of wild mushrooms have been found to have considerable levels of radiocesium in Finland (Rantavaara, 1987). The results obtained by annual monitoring of radiocesium in mushrooms indicate that the contents of radiocesium decrease mostly only by radioactive decay in the long term. Mushrooms are an important carrier of radiocesium to human diet for many years after fallout. The aim of this study was to examine the efficiency of normal household methods to reduce ^{137}Cs contents in edible mushrooms. This information is needed for making realistic dose assessments of dietary radiocesium. Cesium is a highly soluble ion and tends to concentrate in the aqueous fractions during food processing. Many generally used household methods in cooking or preserving mushrooms include soaking or boiling with water.

Material and methods

The mushroom species selected for testing the decrease of ^{137}Cs contents were*: Cantharellus tubaeformis, Cantharellus cibarius, Lactarius rufus, Russula paludosa, Craterellus cornucopioides, Scutiger ovinus, Hydnum rufescens, Hydnum repandum* and salted miscellaneous mushrooms (mainly *Lactarius spp.*). These mushrooms are known to accumulate cesium intensively, and they are consumed widely in Finland. Parboiling is generally recommended to *Lactarius* Spp. before culinary preparation. Drying is a common way to store mushrooms like *Craterellus cornucopioides* and *Cantharellus tubaeformis*, and they are soaked before cooking. Salting is a general method to preserve mushrooms, and the salted mushrooms are either soaked or cooked before use to reduce the salt in them. The methods of parboiling and soaking were based on recipes, which are in general household use in Finland, given by the Martha Organization (a Home Economics Extension Organization). Most of the tests were made with two or more parallel samples.

Soaking of fresh mushrooms. Fresh mushrooms were cleaned and rinsed. Soaking of fresh mushrooms was tested with cold and hot water, soaking times of one hour and over night.

Freezing of mushrooms. Cleaned mushrooms were fry-boiled before freezing. After melting the mushrooms the liquid was sieved and discarded.

Parboiling of fresh mushrooms. Cleaned mushrooms were parboiled starting with different amounts of cold and hot water. The original recipe recommends to use 4-5 liters water for 1 kg mushrooms and to start with cold water, cooking time of 4-5 minutes. After cooking mushrooms were rinsed with cold water.

Soaking of dried mushrooms. Mushroom samples were cleaned from leaves, needles and soil, sliced and dried at 40º C. Hot water was used with soaking time of one hour, and cold water, if the soaking time was over night.

Soaking and cooking of salted mushrooms. The salted mushrooms were soaked with cold water over night or boiled for some minutes, which are normal ways to reduce salt before consumption.

Sample and data analysis. Concentrations of ^{137}Cs and ^{40}K in samples were determined with low-background high-resolution gamma spectrometers with HPGe detectors in cylindrical beakers with volumes of 30 ml or 110 ml, both before and after the treatments. The ^{137}Cs contents (Bq/sample) of all the samples were calculated for both the untreated samples and the samples after treatments. The change in the ^{137}Cs contents of mushrooms following the treatments was defined by using retention factor F_r (Long, 1995).

$$F_r = \frac{\textit{total activity of the radionuclide in the processed food (Bq)}}{\textit{total activity of the radionuclide in the raw material (Bq)}}$$

Results and discussion

The ^{137}Cs concentrations of the unprocessed fresh mushrooms were in most samples 100-200 Bq kg^{-1}. The highest values were measured for salted mushrooms, up to 1000 Bq kg^{-1} (fresh weight). The uncertainties of the gamma measurements for ^{137}Cs were 3-6%.

The ^{137}Cs contamination of the mushrooms used in treatments had come via root uptake, because the samples were taken in 2003-2004. Rinsing of mushroom samples with cold water did not remove more than 10% of the ^{137}Cs contents. The soaking, parboiling and cooking were efficient to remove most of the contamination, approximately 70-80%. The removal of ^{137}Cs was generally increased with larger volumes of water and longer treatment times. The efficiency of the treatments increased considerably if mushrooms were soaked or parboiled twice. Soaking of dried mushrooms with cold water over night was found slightly more efficient than one hour soaking with boiling water. The methods used and retention factors F_r obtained for the treatments are given in Tables 1-3.

Considerable reduction in dietary ^{137}Cs through mushrooms may be achieved with the normal household methods. The reduction is based on discarding the water (liquid) fraction, which is not always done with dried mushrooms in households.

Table 1. *Cantharellus tubaeformis*

Treatment	Treatment time	Sample g/water dl	F_r
Soaking of fresh mushrooms			
-cold water	over night	100 g/5 dl	0.30-0.33
-cold water, twice	2 hours+over night	100 g/5 dl+5 dl	0.13-0.20
-boiling water	1 hour	130 g/2 l	0.05-0.06
Parboiling of fresh mushrooms			
-starting with cold water	until boiling	100 g/5-10 dl	0.21-0.25
-starting with boiling water	5 minutes	100 g/5-10 dl	0.14-0.18
Freezing of fresh mushrooms		200 g	0.26-0.27
Soaking of dried mushrooms			
-boiling water	1 hour	15 g /2 dl	0.35-0.41
“	“	15 g/3 dl	0.27-0.31
“	“	15 g/4 dl	0.28-0.29
“	“	15 g/5 dl	0.26-0.26
“	“	5 g/3 dl	0.13-0.14
-cold water	“	20 g/2 dl	0.21
“	“	20 g/3 dl	0.14
“	“	5 g/3 dl	0.11-0.12
-cold water	over night	15 g /2 dl	0.32-0.33
“	“	15 g/3 dl	0.24-0.30
“	“	15 g/4 dl	0.24-0.24
“	“	15 g/5 dl	0.15-0.22

Table 2. Salted mushrooms.

Treatment	Treatment time	Sample g / water l	F_r
Soaking of salted mushrooms			
cold water, once	over night	200-300 g/1 l	0.15-0.18
cold water, once	over night	200g/1.2 l	0.13
cold water, twice	2 hours+over night	300g/1 l+1 l	0.05
Boiling of salted mushrooms	2-5 minutes	200-300 g/1 l	0.18-0.29

Table 3. Miscellaneous mushrooms.

Treatment/ species	Treatment time	Sample g / water dl	F_r
Soaking of dried mushrooms with cold water			
Craterellus cornucopioides	1 hour	15 g/4 dl	0.31
Scutiger ovinus	1 hour	50 g/8 dl	0.34
Soaking of fresh mushrooms with cold water			
Russula paludosa	1 hour	250 g/40 dl	0.79
Soaking of fresh mushrooms with boiling water			
Hydnum rufescens	1 hour	80 g/14 dl	0.07
Hydnum repandum	1 hour	200 g/24 dl	0.14
Russula paludosa	1 hour	300 g/40 dl	0.06
Cantharellus cibarius	1 hour	50 g/12 dl	0.13
Parboiling of fresh mushrooms			
Lactarius rufus - starting with cold water	5 minutes	400 g/40 dl	0.18
Craterellus cornucopioides			
-starting with boiling water	until boiling	400 g/10 dl	0.32
-boiling twice	1+10 minutes	400 g/10+10 dl	0.16

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Control of long-term transfer of ^{137}Cs to grass and cereal grain in Swedish Chernobyl affected areas, 1986 - 2003

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Abstract: In 1986 the ^{137}Cs fallout from the Chernobyl accident was relatively high in three counties of North Sweden and in two counties of East-Mid Sweden, designated X, Y, Z and U/C respectively. In these counties an investigation of ^{137}Cs fallout comprised a large number of farms, most of the farms on grass site and some on cereal site. The ground deposition of ^{137}Cs varied largely, and was on average higher in X-county than in Y- and U/C-counties and lowest on Z-county. Transfer of ^{137}Cs to crops was determined every year until 1996, and later every third year up to 2003. The transfer of ^{137}Cs was higher to pasture grass than to ley grass and much lower to cereal grain. For all the three crops the transfer was higher on organic soils than on mineral soils. K-fertilization decreased the soil-to-plant transfer of ^{137}Cs as did ploughing and harrowing the ley grass and cereal sites.

Introduction

The Chernobyl accident in April 1986 caused substantial radioactive fallout over a large part of continental Europe and also the Scandinavian countries. In Sweden it was soon decided to start investigations on the transfer of ^{137}Cs to grass and cereal crops on a large number of field sites. The districts selected were parts of the three northern counties, Gävleborg (X), Jämtland (Y) and Västernorrland (Z) and of the two mid eastern counties, Västmanland and Upland (C/U). As determined by inflight measurements by air-crafts, Edvarsson (1991), these counties were reported to suffer from high contamination levels of ^{137}Cs.

As a first step, we investigated the ground deposition, on a grass site and on a nearby arable site and also the interception of ^{137}Cs by grass crops. Studies for both types of crops were intend to continue for a number of years, and at least cover the period of an expected lag period for natural pastures. Detailed reports were given for a time period, of 8 years for the X county and Z- county, Rosén (1996); Rosén, Eriksson & Haak, (1996), and of 10 years for the Y county, Rosén, Haak & Eriksson (1998).

It was later decided to extend the study on some selected sites up to the year 2003, Carlsson (2004). The increased knowledge obtained was regarded as essential for planning and dimensioning agricultural countermeasures after radioactive fallout similar to that of Chernobyl. The transfer of ^{137}Cs to ley grass and pasture grass was compared with that of cereal grain.

The aim of this article is to summarise the results obtained during the time period of 17 years, and especially to discuss the long-term trends in transfer of ^{137}Cs to ley grass, pasture grass and cereal grain in different regions.

Material and methods

Description of farms and sites: The intensity of production and cultivation differed between the farms. In the X, Y and Z counties farming mainly relied on animal husbandry, dairy cows for milk production, sheep and sometimes a small number of goats for meat production. Many

of the farms practised grass and hay production and use of natural pastures, extensive cultivation and only use of animal manure as fertiliser. Other farms practised more intensive cultivation in crop rotations with grassing leys and apply mineral fertilisation. In the C/U-counties animal husbandry was less frequent and most of the farms practised intensive cereal production.

The prerequisite for the sites to be sampled was, that they should be on farms with problems due to high ^{137}Cs content in pasture grass and hay in 1986. Furthermore the sites should be situated at least about 100 m from the buildings, large trees providing shelter, or the edge of the forest. One site was natural pasture or grassing ley and the other ploughed land to be sown with cereals and other crops. The two different sites were situated on adjacent fields. Both the sites consisted of a level land area to be used for random sampling of soil and crops.

Both peat soils and different mineral soils were investigated, Table 1. As shown the numbers and distribution of different types of soils varied between the counties.

Table 1. Number of peat soils and number and type mineral soils

County	Total num-ber of soils	Number of peat soils	Number and type of mineral soils			
			Sand	Loam	Silt	Clay
C/U county	14	5	1	4	1	3
X county	16	8	2	6	-	-
Y county	32	-	4	23	4	1
Z county	13	1	7	5	-	-

Sampling methods: The soil was sampled by taking three to four soil cores from different soil layers 0-5 or 0-10 cm. The core samples from each layer were pooled to a bulk sample, air-dried at 30^{o}C for 1 week, weighed, ground and passed through a 2-mm sieve. The bulk samples were then used for classification of soils and chemical soil analyses, for determination of the ground deposition of ^{137}Cs, the latter calculated as an integrated sum over layers down to uncontaminated soil depth, thereby corrected for surface and volume of each soil layer, Rosén (1996).

The sampling of the crops comprised 2-4 micro-plots of 0,25 m^2 at normal harvest times of the year. Mostly the grass was harvested twice and cereals once per year. The grass and cereal straw was cut about 5 cm above ground. The four sub-samples were quantitatively pooled to bulk samples, which then contained the yield from the area of 0.5 to 1 m^2. The bulk samples of crop material were dried at 90 ^{o}C.

Analytical methods: The activity concentrations of ^{137}Cs, Bq/kg dry weight (d.w.), in plant and soil samples were determined by means of a high purity computer-aided germanium detector system, housed in a low background laboratory. The measurement errors were in the range of 1-10 percent or 24 hour. All activity concentrations in soil samples were recalculated to the ground deposition of ^{137}Cs, kBq/m^2, at 1 May 1986. To be able to compare results of activity measurements in plant and soil from different sites and dates of sampling, a transfer factor was determined. This requires for a specific site and sampling date that the activity concentrations of ^{137}Cs in plant, has to be recalculated to the actual sampling date of the crop sample. The unit, which is a concentration ratio, has been denoted TF_g, IAEA (1987), and is recapitulated here as;

$$TF_g = \frac{\text{activity concentration in plant (Bq/kg d.w.)}}{\text{activity deposited on ground (Bq/m2)}} \qquad \text{Unit} = m^2/\text{kg d.w.}$$

Results and discussion

Transfer of ^{137}Cs to crop products: The transfer of ^{137}Cs to pasture grass, ley grass and cereal grain, is given as mean values of TF_g for each county and for five years, in Tables 3-5. The general trend was that the transfer decreased with year, and relatively more from the fallout year 1986 to the next year 1987 than successively during the following years up to the year 2003.

The mean ground deposition of ^{137}Cs was lowest in Z-county and highest in X-county as shown in the Table 2 below.

Table 2. Ground deposition on agricultural land, kBq/m^2, at 1 May 1986, in the most Chernobyl affected counties of Sweden, as expressed by different statistical measures

County symbol	Number of soils	Means of values	Range of values	Standard deviation	Coefficient of variation
C/U	14	58.1	2 – 98	26.8	46.1
X	16	125.8	20 – 203	58.8	46.7
Y	32	43.3	14 – 88	21.9	50.6
Z	13	31.4	17 – 44	8.3	26.4

Comparison of Table 3 with Table 4 shows that TF_g in X- and Z-county was several times higher for pasture grass than for ley grass. This difference reflects the higher sensitivity for ^{137}Cs transfer to grass on uncultivated pasture sites than on cultivated and fertilised ley sites. Otherwise the time trends of decrease with years were about the same.

Table 3. Mean transfer of ^{137}Cs to pasture grass, TF_g (m_2/kg d.w.) x 10^{-3}, in 1986 –2003

Coun-ty	1986		1987		1990		1994		2003	
	TF_g	(n)	TF_g	(n)	TF_g	(n)	TF_g	(n)	TF_g	(n)
C/U	41.8	(3)	23.7	(4)	2.7	(3)	3.9	(3)	1.1	(4)
X	83.4	(5)	74.0	(4)	10.6	(5)	9.5	(4)	7.5	(6)
Y	53.1	(5)	26.8	(5)	1.3	(4)	0.9	(2)	1.4	(5)
Z	133.5	(6)	57.3	(5)	36.4	(4)	27.0	(4)	7.5	(5)
Mean	**84.7**	(19)	**45.1**	(18)	**13.2**	(16)	**12.3**	(13)	**4.7**	(20)

Table 4 shows that the transfer of ^{137}Cs to ley grass decreased differently with time. In 1986 the transfer was highest in Z-county and lowest in C/U-counties, in 1987 highest in Y-county and lowest in C/U-counties. Later TF_g decreased relatively less in X-county and Z-county than in C/U-counties and Y-county.

Table 4. Mean transfer of ^{137}Cs to ley grass, TF_g (m_2/kg d.w.) x 10^{-3}, in 1986 –2003

County (symbol)	1986		1987		1990		1994		2003	
	TF_g	(n)	TF_g	(n)	TF_g	(n)	TF_g	(n)	TF_g	(n)
C/U	13.4	(8)	8.5	(10)	-		-		0.1	(9)
X	75.7	(7)	22.4	(8)	4.7	(10)	1.6	(6)	1.0	(7)
Y	28.4	(25)	6.3	(27)	0.6	(19)	0.5	(8)	0.3	(24)
Z	55.9	(5)	24.2	(5)	11.8	(4)	8.9	(7)	2.4	(7)
Mean	**36.1**	(45)	**27.8**	(50)	**3.2**	(33)	**3.6**	(21)	**0.7**	(47)

Cereal grain in Table 5 shows that the transfer of ^{137}Cs to cereal grain in X- and Y-counties was much lower than that in grass. The transfer to cereal grain was lower already in the fallout year 1986 and remained at a very low level up to 2003. In 1986 the uptake of ^{137}Cs in cereal grain was 20-45 and 25-50 times lower than for ley grass and pasture grass.

Table 5. Mean transfer of ^{137}Cs to cereal grain, TF_g (m_2/kg d.w.) x 10^{-3}, in 1986 –2003

County	1986		1987		1990		1994		2003	
	TF_g	(n)	TF_g	(n)	TF_g	(n)	TF_g	(n)	TF_g	(n)
C/U	-		-		-		-		0.04	(4)
X	4.24	(2)	0.09	(2)	-		-		0.03	(2)
Y	0.56	(10)	0.28	(1)	0.19	(7)	0.26	(1)	0.04	(4)
Z	-		-		-		-		-	
Mean	**1.17**	(12)	**0.15**	(3)	**0.19**	(7)	**0.26**	(1)	**0.04**	(10)

Conclusion

The study shows that the transfer of ^{137}Cs to grass decreased with time, and most from the fallout year to the next year 1987.

The transfer was higher to pasture grass than to ley grass and lowest to cereal grain. This depends on higher ^{137}Cs fixation to mineral soil, ploughing and to fertilizing with potassium in cultivated grassland, ley grass and cereal grain site, than in uncultivated grassland, pasture site.

The transfer to grass was highest in the Z-county during the study time despite that this county had the lowest deposition.

The uptakes of ^{137}Cs, after 1990 up to 2003, to different crops and on different site are more or less stabilized on different levels.

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Malicious use of radioactive material - are we prepared for the unknown?

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Abstract: Lately there has been much concern that terrorists may use nuclear devices or radioactive material. The most dangerous threat would be theft or diversion of fissile material followed by its use in an improvised nuclear device or the use of a complete weapon. Radioisotopes dispersed by conventional explosives or fire is also a possibility, but this method cannot bring about large-scale devastation like a nuclear device. It could, however, still bring fear to society and cause socio-economic disruption. It is not easy to stage a radiological attack and the effects would be difficult to predict. This may be the reason why such events have not yet occurred, but a radiological terrorist attack cannot be excluded in the future. The problem for the society is that first responders generally have no experience of this situation in real life as ”every-day accidents” involving radioactive material almost never happen. Radiation monitoring equipment and special training is needed to cope with a malicious event. In the long term it is a challenge for a response organisation to maintain enough serviceable equipment and skill to handle a possible future nuclear or radiological terrorist attack.

Introduction

The use of radiation sources in nuclear industry, medicine and technology is subject to extensive regulatory control. Highly radioactive sources should be kept safe and secure. Accidents with radiation sources involving the public are not common, but despite all safety precautions accidents sometimes happen. In the 40-year period 1962 - 2001 over 20 radiation accidents have occurred worldwide with orphan (uncontrolled) sources causing 28 fatalities in the public. 21 of these fatalities happened in the last 20-year period at 16 different occasions (accidents caused by X-rays, accelerators, medical treatment and reactors or critical assemblies are not included). IAEA (1998, 2002). Severe cases happened in Morocco and Brazil. In 1984 in Morocco an entire family of 8 was eradicated by a lost Ir-192 radiography source that was picked up by a passer-by unaware of the danger, UNSCEAR (1993). In 1987, 249 people were contaminated in Goiânia in Brazil by Cs-137 dispersed from a stolen teletherapy source. There were 4 fatalities and 112000 people were monitored for suspected contamination. The human consequences were substantial and the society suffered large economic losses due to the fear of radiation, IAEA (1988).

In the last few years an additional threat has emerged in the sense that evil-doers or terrorists may want to use radioactive material to cause injury to people and disrupt society. A device called a radiological or ”dirty” bomb could for example do this. Compared to a conventional bomb there is an additional prolonged effect from the radioactive material with the potential for physical injury and death to persons who were not wounded in the initial blast. Radiation effects take days and weeks to manifest. This slowly showing ”invisible” danger could bring fear to society and cause socio-economic disruption. The world has not yet witnessed any terrorist attacks involving radioactive material, probably due to the difficulty to get hold of and handle this type of material. The possibility, how-

ever, that a potential perpetrator will be able to obtain enough material to stage some kind of radiological attack cannot be ruled out. The question is - are we prepared to handle this situation?

Prerequisites for a radiological attack

Motive. Radioactive material is a ”slow killer” compared to more fast-working means such as guns, explosives, fires or poisons that evil-doers have applied during centuries. A potential evil-doer or terrorist must have a reason why he or she wants to use radioactive material, since the effects (except for nuclear explosives) will be delayed and therefore uncertain because of the time for countermeasures from the society. One motive could be to create fear that may arise in the public from the invisible radiation threat. This could affect thousands or hundred of thousands of people, as in the case in the Goiânia accident. Another motive could be the ”first-time” effect. The person or organisation that will pull off the radiological attack will most surely attain large attention. The most extreme attack is the detonation of a device with even a small nuclear yield in a metropolitan area. Such an attack could cause immense casualties and cause enormous human, social and economical devastation. It could even lead to changes in political systems.

Knowledge. Handling highly radioactive material requires at least fundamental knowledge in radiation protection. Even if a potential perpetrator is willing to sacrifice his own life it will be necessary to avoid early injuries (such as severe radiation sickness) that could incapacitate the perpetrator before the task is done. The handling of fissionable material in quantities needed to produce critical masses for nuclear fission is extremely dangerous and advanced knowledge in nuclear physics is necessary. The records of criticality accidents show this. Since 1945, 60 criticality accidents or nuclear chain reaction excursions have occurred in the nuclear industry, weapons production facilities and laboratories, Los Alamos (2000).

For highly radioactive material there is a need to construct a dispersion device or find out a way to deploy the radioactive material to cause harm. Explosives, fire or mechanical means could do it. It is not, however so easy to disperse a radioactive material efficiently. The outcome will depend on its physical and chemical form. To be sure that the ”right” effect is obtained, some tests will probably have to be performed with the actual material. This will take time and could lead to unveiling of the plans.

Material. Radioactive material in enough quantities to cause harm cannot be easily acquired on the market or produced at home from raw material. Fissile material, Pu-239 and U-235 in highly enriched uranium (HEU) are virtually impossible for a single person to obtain in enough quantities to produce a nuclear yield. It cannot be excluded, however, that a terrorist organisation with international network could get hold of fissile material. Many of the worlds research reactors use HEU as fuel. Another way would be to steal a complete stockpiled weapon, but safeguards in the device may prevent unauthorised parties to trigger a detonation.

Radioactive sources (radioisotopes) for the industry or medical applications are produced in many research reactors. Russia has the highest number of research reactors followed by USA, Japan, France, Germany and China.. Many different radioisotopes are produced for various purposes. Those with highest potential to cause harm are Co-60 ($T_{1/2}$ 5.3 years) Sr-90 ($T_{1/2}$ 29 years), Cs-137 ($T_{1/2}$ 30 years) and Ir-192 ($T_{1/2}$ 74 days). These have caused many of the severe radiation accidents that have affected the public. High activity sources

are not impossible to steal, but could pose a severe danger for the person handling the source. A transport of Ir-192 from a production reactor to an equipment factory for radiographic devices can typically contain 370 TBq, which will give a deadly radiation dose within minutes at 1 metre distance if the source is opened or taken out from its heavy shielding. Radioisotopes used in medicine for diagnostic purposes such as I-131 ($T_{1/2}$ 8 days) and Mo-99/Tc-99m ($T_{1/2}$ 2.75 days) have short half-lives and are shipped each week. This gives rise to a large number of transports. Transportation rules, however, restrict the activity of each package to 0.7 TBq for I-131and 4 TBq for Mo-99/Tc-99m. This limits the potential for the source to cause widespread harm even if taken out of its shield and dispersed.

Possible scenarios

Nuclear devices. Theft of a complete weapon, theft of fissile material and the making of a primitive device can lead to enormous consequences in the hands of a competent terrorist organisation. The probability may not be high because of the huge difficulties to get hold of fissile material and construct fissile devices with even a small yield, but the effects would be disastrous. Detonation of a nuclear device produces an extremely hot fireball. The radiant energy is enough to ignite material and cause burns far from ”ground zero”. Even for a limited yield of 0.1 kT the shock wave will cause 50 percent fatalities at 130 metres and thermal burns at 200 metres. The initial radiation during the first second will cause 50 percent fatalities out to 460 metres (4 Gy). The residual radiation from early fallout will produce the same dose out to about 2800 metres in the downwind direction during the first hour to people who are not sheltered. NCRP (2001). Even this very limited nuclear yield could cause tens of thousands of deaths if the device is detonated in a city. Because of the high radiation levels after the detonation it will be impossible for rescue teams to enter the near area in time to help those who have survived the initial blast.

Nuclear industry. Serious consequences could result from an attack on a nuclear power station because of its large amount of radioactive material. A number of technical safety systems are built into a nuclear power plant to prevent large releases. An evil-doer or terrorist has to take out or circumvent the safety systems to create a large release. In the worst case it could lead to fatalities in the first 5 - 10 kilometres downwind for people who are not sheltered. Widespread long-term land contamination and food contamination could result. An attack on a nuclear fuel factory with low enriched uranium (LEU), an attack on a nuclear fuel transport or theft of nuclear fuel could lead to events with criticality and high radiation levels within a few hundred metres from the event. But it will not be possible to construct a device with nuclear yield from LEU.

Radioactive substances. Theft of radioactive material and dispersion of the material to cause human exposure is a terrorist scenario that can happen in the future. A large number of radiation sources exist that could fall in the hands of an evil-doer. To reach an effect, however, a perpetrator has to construct some kind of device for the dispersion. Conventional explosives together with a highly radioactive source would be one way. Another way would be to use heat or fire, but the effect would be very unpredictable. Radioactive material such as Co-60 and Ir-192 is in metal form and would probably not disperse very well or maybe not disperse at all. Other materials such as Sr-90 (rear earth) or Cs-137 (alkali metal) are probably easier to disperse, but the final effect is still unpredictable. The area that can be affected is limited. Depending on the amount of activity it

could be a few hundred square metres or up to a few square kilometres. People within the area can get severe radiation burns if they come in contact with radioactive particles. Contamination of skin, hair and clothing is likely, leading to immediate need of personal monitoring and decontamination. The demand for medical help could be huge. The delayed effects of radiation could worsen the situation as many people, who are not directly affected still would search medical help.

The difficulties of emergency preparedness. Conclusions

Radiation accidents are rare and ”every day accidents” involving radioactive material (almost) never happen. Therefore, rescue service personnel will not obtain practical experience from radiation threats in their daily work. There are also generally long periods between exercises for the average fireman or policeman when it comes to nuclear or radiological emergencies (except maybe in the emergency planning zones of nuclear installations). Radiation monitoring equipment must be maintained and kept in good repair, but in the long term it may by difficult to keep the equipment serviceable and easy at hand when needed.

It is not a straightforward task for a terrorist to stage a nuclear or radiological attack, but if an attack eventually will occur it could cause fear and put society into hardships. The uncertainty of what has happened will most probably be high in the beginning and the full effect of the attack not seen until days or weeks after the initial event because of the delayed effect of radiation injuries. To cope with these situations it will be necessary to maintain continuing education and training of first responders and to keep special task teams that can solve the unknown questions.

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^{137}Cs accumulation in coastal sediments in Sweden

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Abstract: Seabed sediment samples were collected in 1998, 2000 and 2001 at 20 sites located in the Baltic Sea and 4 sites in the Skagerrak. The objectives of the sampling campaigns were (i) to establish the coastal sediment distribution of ^{137}Cs, (ii) to evaluate the vertical core distribution of ^{137}Cs, (iii) to study the sediment accumulation rates, and (iv) to assess the sediment inventories of ^{137}Cs. The results show a very high variation in ^{137}Cs concentrations and an almost 100-fold difference in inventories, showing predominance of Chernobyl derived ^{137}Cs in the Baltic Proper compared to the western Baltic and the Skagerrak areas. Sediment accumulation rates were highly dependent on sediment types and ranged from 0.05 to 1.8 cm y^{-1}.

Introduction

Seabed sediment samples were collected during three cruise expeditions along the Swedish coast in June 1998 (GAUSS #320), July/August 2000 (GAUSS #352) and in June/July 2001 (GAUSS #369), (Fig. 1), organized by the German Federal Maritime and Hydrographic Agency. Six sites located in the Baltic Proper and 2 sites close to the Danish Straits were investigated in the 1st cruise. Five sites located in the Bothnian Sea of the Baltic Sea and 3 sites in the Baltic Proper were investigated in the 2nd cruise. In the 3rd cruise 8 sites were investigated, 1 site located on the southern part of the Baltic Proper, 3 sites located on the western coast in the Kattegat, and 4 sites situated in the Skagerrak.

The objectives of the sampling campaigns were (i) to establish the coastal sediment distribution of ^{137}Cs, (ii) to evaluate the vertical core distribution of ^{137}Cs, (iii) to study the sediment accumulation rates, and (iv) to assess the sediment inventories of ^{137}Cs.

Material and methods

The sediment samples were collected by means of gravity corers and box-core samplers (sub sampling with cylindrical tubes) for sediment depths down to about 50 cm. The sediment cores were sliced on-board at 1 cm intervals and collected and sealed in plastic bottles. After freeze-drying in the home laboratory and determination of sample wet- and dry weights, the samples were homogenized and transferred to standard geometry plastic tubs (60ml, 90 ml volume) for analysis by gamma spectrometry in a low-level underground laboratory. The sediment concentrations of ^{137}Cs and ^{226}Ra were assessed by using ^{137}Cs and ^{226}Ra standard solutions (traceable to NIST) thoroughly mixed in inactive sediment materials.

The gamma spectrometry analysis were followed by radiochemical analysis of ^{210}Po. About 0.5-2 g dry sample material was used together with ^{208}Po standard solution for

yield determination. The samples were wet ashed by microwave treatment using a standard protocol at approx 200°C/12 bar. After dissolution and conversion into chloride form, Po was deposited spontaneously onto Ag discs using a standard protocol. The ^{208}Po and ^{210}Po content was assessed by alpha spectrometry using a Ortec Octete^{TM} system with PIPS detectors. The sediment accumulation of ^{210}Pb was then obtained by assessing the non-(^{226}Ra)-supported ^{210}Pb concentrations using the ^{210}Po and ^{226}Ra data. The calculations were made from the assumption of constant rate of supply (CRS) of ^{210}Pb.

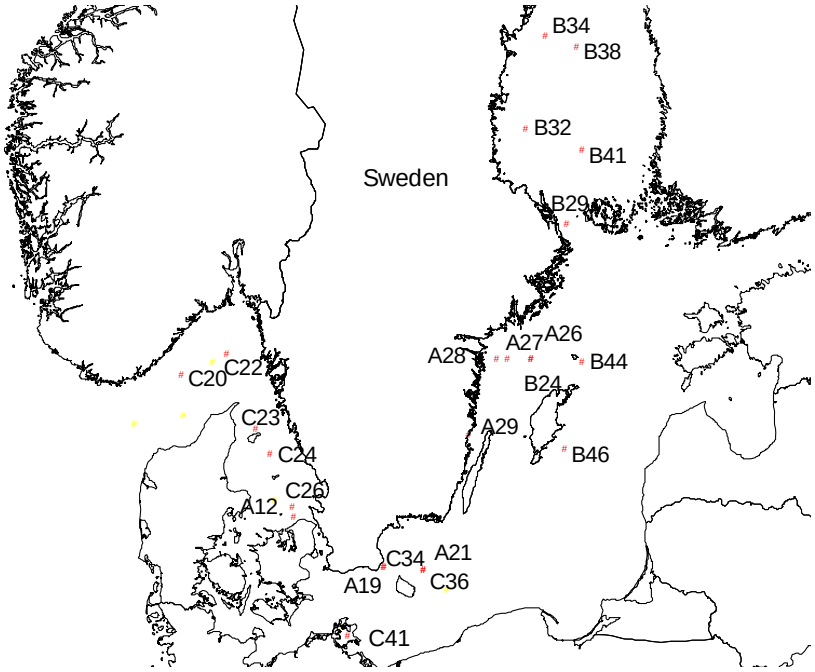

FIG. 1. Sampling sites

Results and discussion

The set of graphs in Fig. 2 show the depth distributions of ^{137}Cs and ^{210}Pb for 6 of the 24 sampling sites (site symbols A denote the 1998 cruise, B the 2000 cruise and C the 2001 cruise). The sediment depth distributions show in general a typical distinct peak concentration, ranging from parts of cm up to about 30 cm, followed by a monotone decline with depth. The depth of the mixing zone is much dependent on the sedimentation rate and organic content. In general, the maximum ^{137}Cs concentrations are at least one magnitude higher in the Baltic Sea compared to the Skagerrak and Kattegat areas, showing the predominance of deposition from the Chernobyl accident release. The total input to the Baltic Sea from the Chernobyl accident amounts to about 4700 TBq, compared to the estimated deposition of 900 TBq from atmospheric weapons tests and about 250 TBq from European nuclear reprocessing releases (EC 1998).

The sediment accumulation rates (Fig. 2) show on large variation, ranging from 0.05 to 1.8 cm y^{-1}. The sediments are of different types; sands, sandy mud, muddy sand, varved clays, pelitic muds, fine aleurite muds and moraine deposits. In addition, the organic content in the upper part of the sediments showed on large variation. No clear correlation between accumulation rates and sediment type was observed. Neither is there an indica-

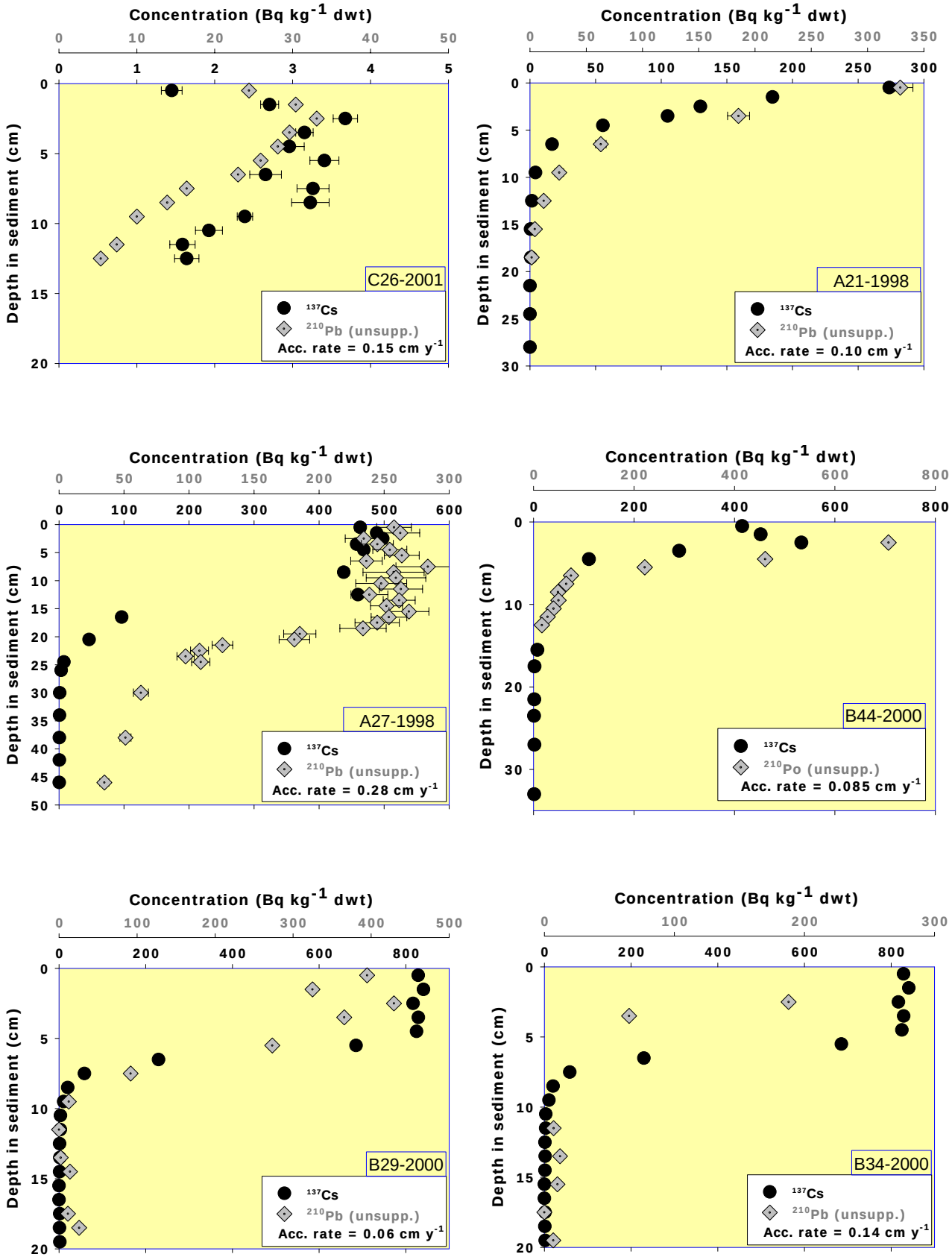

FIG. 2. Concentrations of ^{137}Cs and ^{210}Pb and sediment accumulations rates in Baltic Sea sediments

tion of expectedly higher deposition rates in deep basin areas or with water depth in general. However, the deposition rates show good agreement with literature data for the Baltic Sea (Jonsson et al. 1990, Jonsson 1992) and for the Gulf of Finland (Vallius et al. 1998).

The sediment inventories of ^{137}Cs are displayed in the figure 3, showing an almost 100-fold difference in inventories. In general, as expected, the western coast display very moderate inventories, comparable to the cumulative deposition from global atmospheric fallout, whereas the inventories in the Baltic Sea show a predominance from the Chernobyl accident release (for some sites comparable to the terrestrial deposition of the areas).

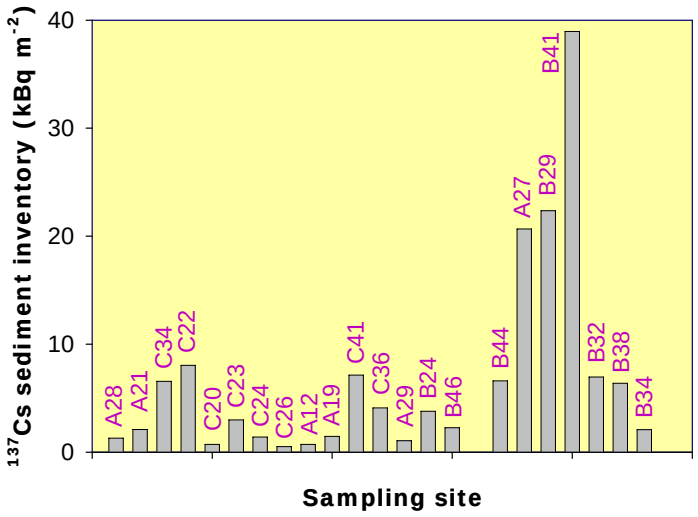

FIG. 3. ^{137}Cs sediment column inventories in the Baltic Sea

Acknowledgements

The assistance from the R/V Gauss staff during the three expeditions in sampling, providing hydrological data and for the ship arrangements is greatly acknowledged. The work was financially supported by the Swedish Radiation Protection Authority.

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Session XIV: Regulatory and International Activities

Inspections by the Swedish Radiation Protection Authority

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Abstract: The Swedish Radiation Protection Authority, SSI, is the regulatory and supervisory authority in Sweden that is responsible for radiation protection on a national level. SSI employs 110 professionals with a broad and varied expertise; engineers, physicists, chemists, legal experts and press officers and is headed by a government-appointed director general. The budget is about 10 million € per year and is financed by taxes and fees. SSI is divided in to five different departments. This paper will describe mainly the inspection activities performed by the Department of Occupational and Medical Exposures at SSI.

The use of ionising radiation in Sweden

The use of ionising radiation in Sweden is quite comprehensive. We have four nuclear power plants with a total of 12 reactors, a research reactor site, a fuel factory, facilities for waste and spent fuel, more than 1000 non-nuclear industries with licensed equipment, 11 major sites with research facilities, 232 hospitals performing x-ray diagnostics, nuclear medicine, radiation therapy and research, 475 dentists with individual and 12 000 dentists with general licenses and finely 264 veterinaries with licenses for the use of x-ray equipment. Supervision at the Department of Occupational and Medical Exposure has been divided into tree major areas: nuclear installations, industry and research, and medical exposures.

Nuclear installations: The nuclear industry in Sweden comprises 10 nuclear power reactors in operation and two that are closed down, one research reactor, one nuclear fuel factory, one final repository for radioactive waste, and one central interim storage facility for spent nuclear fuel. SSI performs inspections of all the nuclear facilities several times a year, covering occupational exposures as well as effluents and waste handling.

Industry and research: Within the broad sector of non-nuclear industry and research activities outside the medical sector the authority supervise the use of open and sealed sources as well as x-rays and accelerators in various applications. There are about 6000 radiation sources at 1500 licence holders altogether. The selection of inspection objects is based on the assessment of priorities for the licensed activities, which is taken into account risk factors such as high dose rate, high activity, or complex use of sources.

Medical exposures: Within the medical sector there are 232 hospitals/medical facilities with x-ray equipment, where 5 million examinations per year are performed. In the 33 hospitals with nuclear medicine departments 120 000 examinations and 4000 treatments are performed every year. In 17 hospitals with external radiation therapy departments 20 000 treatments are performed yearly.

In Sweden there is a system with a general license that allows registered dentists to use x-ray equipment intended for intra-oral image receptor with a tube potential up to 75 kV. There are approximately 12 000 registered dentists in Sweden.

There are 264 veterinarians with a license to use x-ray equipment, three with a license for nuclear medicine and two for radiation therapy with I-131.

Legal framework

The basis of the legal framework is the "Swedish Radiation Protection Act" from 1988. The act is requiring a licence for all work with ionising radiation and general radiation protection obligations are given. All activities with radiation are covered in the act that is a basic frame law; meaning that it is dealing with general issues and have to be complemented with additional, more detailed regulations. According to the European basic safety standard, 96/29 Euratom, SSI has issued a number of regulations. The regulations are stressing the importance of quality assurance programmes for all activities involving ionising radiation. The regulations can be downloaded in English from our web site: http://www.ssi.se.

Authority

The Swedish Radiation Protection Ordinance is assigning SSI as the competent authority with the task of: issuing regulations, granting licences, performing supervision including inspections and promoting research. SSI has the legal authority to decide what action should be taken by the licence holder in order to improve radiation protection by issuing prescriptions and prohibitions. SSI can also prescribe a penalty for non-compliance or withdraw a licence.

Inspection Policy with Strategy and Guidelines for Inspection

In recent years, SSI has improved its policy for inspections in order to create an updated and uniform programme for all areas and activities supervised by SSI.

The inspection policy describes the approach to and goals for SSI's inspection activities:

- Inspection is a main task for SSI
- Inspections can be conducted in all areas involving radiation
- Inspections are to be conducted where they accomplish most benefit
- Inspections are to be conducted with the right level of quality and in accordance with the adopted steering documents
- Inspections are to be characterised by integrity and professional behaviour.

The strategy specifies how SSI should work in order to meet policy goals:

- Inspection is a high-priority method of supervision
- Severe accidents and incidents should be immediately followed by an inspection
- The inspections are to be based on SSI's priorities.

High-priority facilities for supervision according to the policy are:

- Those that result in or can result in high radiation doses
- Those where there is a considerable risk that many can receive minor but not negligible radiation doses
- Those about which SSI needs to have more knowledge.

The inspection process: Of the 110 employees at SSI approximately 20 are involved in supervision and inspections of occupational exposure. The frequency of inspections of different objects is varying considerably. Table 1. Dentists with a general license for x-ray equipment are inspected only occasionally. 20 years ago 400 of the approximately

12 000 dentists with a general license were inspected. For 2006 we are planning to inspect 400-600 randomly selected dentists. With 30 inspections per year in the medical field university hospitals and other major hospitals will be inspected every third to fifth year and minor hospitals every fifth to tenth year.

Major research facilities and pharmaceutical industries are inspected every fifth year or when significant changes have accrued. Industrial radiography companies and other holders of strong sources are regularly inspected in a similar way. Industries with simple gauges and similar equipment are seldom, if ever inspected. For the latter group of licensees there is an administrative check for the correctness of the information about registered sources. Disused sources sent to recognised facilities are followed up by information in the register. In addition information about imported and exported radiation sources is collected for statistical flow analysis.

Nuclear facilities are inspected several times per year. The frequency of inspections varies depending on which items are of actual interest but is normally larger for the nuclear power plants compared with the other nuclear facilities. The reason is that the plants are larger facilities and that there is a higher potential for radiation exposures.

Table 1. The inspection frequency

Practices inspected	Number of inspections per year	Number of objects to be inspected	Part inspected per year (%)
Dentists	0	12000	0
Industry	30	1500	2
Veterinarian	5	264	2
Medicine	30	300	10
Nuclear installations	40	6	600

The basis for all inspection activities is our regulation. As a consequence of the European basic safety standard, 96/29 Euratom, SSI has issued a number of regulations. The regulations are stressing the importance of quality assurance programmes for all activities involving ionising radiation. During the inspections we check that the user has all documents, procedures and equipment required according to the regulations. Radiation measurements are performed only occasionally.

The inspection process is a continuous process where inspection experience from previous years is used as the basis for planning inspections for subsequent years. The process is divided into six stages: Selection and Planning–Preparation and Planning–Inspection–Post inspection work–Follow up–Evaluation.

Essentially three types of regulatory inspections are performed:

System Inspections: In a system inspection, the organisation, administrative procedures, organisational co-ordination, allocation of responsibilities and competence at the facility

are examined. The aim of a system inspection is to acquire a sound knowledge of the entire or parts of the quality system of the facility.

Specific Inspections: A specific inspection is focusing on a part of the facility's activity. An event that is important for the radiation protection often initiates a specific inspection. Sometimes, a specific inspection can be initiated by the public or by media. A specific inspection can be part of a follow-up of previous inspections.

Topical Inspection: A topical inspection consists of a number of specific inspections that are performed in co-ordination at a number of similar facilities and with a common theme.

After each inspection a report is written and the licensee is given the opportunity to check all facts in the report before any final decisions are taken on what remedial actions the licensee has to perform. The licensee has to respond in writing verifying that actions have been taken, within a time limit the length of which is depending on the issue. Usually SSI is satisfied with the actions taken by the licensee. Very rarely new decisions under penalty have to be taken.

In addition to decisions SSI gives feedback to the licensee by reports and guidelines. SSI is also organising annual meetings with radiation protection managers at nuclear power plants, with medical physicists and with qualified experts.

Lack of compliance with regulations: The most frequent findings are inadequate education and training. In the medical field most problems are found outside the department of radiology, frequently the education of cardiologists and orthopaedist is inadequate. Inadequate organisation of radiation protection and lack of qualified expert are other frequently found discrepancies.

Some universities have poor local organisations for radiation protection and lack of required qualified expert and lack of RP knowledge at all levels. Large high tech industries usually have well designed organisational structures and clear ambitions to perform well. Findings of weaknesses in radiation rooms and equipment at radiography companies do appear. The need to inform on and to explain the formal regulations is quite common at all contacts with license holders and their employees.

In general, there is a high degree of compliance with the SSI regulations at the nuclear facilities. One reason for that is probably the relatively high frequency of inspections from SSI. In recent years SSI has noted that economic issues is of more importance for the owners than before due to the deregulation of the electric market in 1996. As a consequence SSI today has to be more specific in the regulatory work. There is also important to keep the inspection frequency high in order to ensure that the present radiation protection conditions are maintained.

Conclusions

SSI has high demands on the users to have a quality assurance programme for all activities involving ionising radiation. The programme includes self-assessment, audit, education and training. During inspections SSI is supervising the compliance of the users with laws and regulations by checking documents, procedures and equipment, but rarely by performing measurements. SSI works continuously to improve the quality of supervision and "we do believe that our inspections are justified but not yet optimised".

New methods of performing radiation protection inspections in Norway

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Abstract: In recent years the Norwegian Radiation Protection Authority (NRPA) has put a lot of effort in working out new national radiation protection legislation. Less time-consuming administration of radiation users and radiation sources was one of the goals of the new legislation, hopefully giving more time for inspections. In the same period NRPA has tested out new methods of performing radiation protection inspections, going beyond the traditionally technical inspections. These methods, and also promising methods used by other regulatory bodies in the Health, Security and Environment (HSE) area in Norway, are described in this paper.

Introduction

NRPA has, under different names, been the regulatory authority concerning the use of radiation sources, dosimetry and radiation protection in Norway since 1939. The brief history of NRPA from the earlier technical dosimetry services to today's thematic and system inspections is described in Davidson et al. (2004). In recent years NRPA has put a lot of effort in working out new national radiation protection legislation. Today the radiation protection legislation in Norway consists of the Radiation Protection Act from 2000 (replacing an act from 1938) and the Radiation Protection Regulations from 2003 (replacing regulations from the 1970-ties and 1980-ties). The act and regulations are available in both Norwegian and English at NRPA's web site, www.nrpa.no.

Less time-consuming administration of radiation users and radiation sources was one of the goals of the new legislation, hopefully giving more time for inspections. The Radiation Protection Regulations came into force on 1 January 2004. After that time NRPA has worked out a number of guidances in connection to the Radiation Protection Regulations and new administrative practices has been established. Due to this extra work in connection with introducing the new regulations there has been no extra time released for inspections so far, but new methods of performing radiation protection inspections has been tested out. The new inspection methods go beyond the traditional technical inspections. This paper briefly describe thematic inspections, system audits and random regional inspections as methods, and the experience NRPA and other HSE regulatory bodies in Norway has in using this inspections methods.

Inspection methods

In the first period of NRPA history, dosimetry control in all hospitals having X-ray or radium sources for therapy was the prime activity. From about 1960 to the mid 1980-ties the NRPA inspections were mainly technical in nature. NRPA still perform technical inspections, but today always in combination with checking the radiation protection organisation and systems. Inspection of one to three companies at the same time, still count

for a lot of the inspections performed by NRPA. NRPA has performed thematic inspections for nearly the last two decades, but system audits have been a topic only for the last few years. Inspection methods described in this paper are much the same as the methods used by the Swedish Radiation Protection Authority, referring to Cederlund et al. (2004).

Thematic inspections: In thematic inspections a specific radiation protection problem or equipment is addressed in several companies at the same period. Industrial radiography, blood irradiators, use of unsealed radioactive materials, and interventional radiological procedures are some of the areas where NRPA has performed thematic inspections the past years. In 2004 NRPA performed thematic inspection of all eleven blood irradiators in Norway.

The purpose of thematic inspections is dual. On the one hand it gives a picture of the status for a certain type of radiation use, and on the other hand it result in increased competence at the national authority.

System audits: Before starting a system audit, written instructions, work procedures, logs and other internal documentation of the operator in question are thoroughly examined. During the audit interviews are carried out with several persons on **all** levels in the organisation, combined with on-site verifications if necessary. One of the system audits NRPA performed in 2004 were at the University of Tromsø. Use of unsealed radioactive materials in laboratories and discharge of radioactive substances was the main topics of this audit.

System audits represent a thorough examination of the operators radiation protection system. The radiation protection organisation, dedicated responsibilities, administrative systems, internal cooperation and coordination, competence on different levels in the organisation, technical equipment, ALARA inspired procedures and other relevant radiation protection issues for the practice in question can be evaluated. By the end of the audit all results are presented at an end-meeting. After the audit the operator receive a report including the deviation from the legislation and internal documents revealed during the audit. Then it is the operators duty to find correcting actions, and NRPA can accept or reject the proposed actions.

Random regional inspections: This year NRPA will test out a new inspection method. The intention is to get wider effects of the resources put into the inspections. A letter with announcement of inspections will be sent to all companies in a county or region having a certain type of radiation sources (e.g. industrial radiation gauges), together with updated information on legislation and other relevant information. Only 10 to 25% of the companies will actually be inspected, and the list of the chosen companies will be official only a few days in advance. This inspection method is illustrated in figure 1.

Press release in the local media is commonly used in this inspection method to get focus both on the inspections and the result.

A report with a summary of the results will be distributed to **all** the companies. This method has been used with success by other regulatory bodies in Norway. This inspection method gives all the companies in a county or region updated radiation protection information, and also companies not inspected will in a period put extra focus on the radiation protection system. For the national authority, this inspection method gives a picture of the status for a certain type of radiation use in the county or region in question.

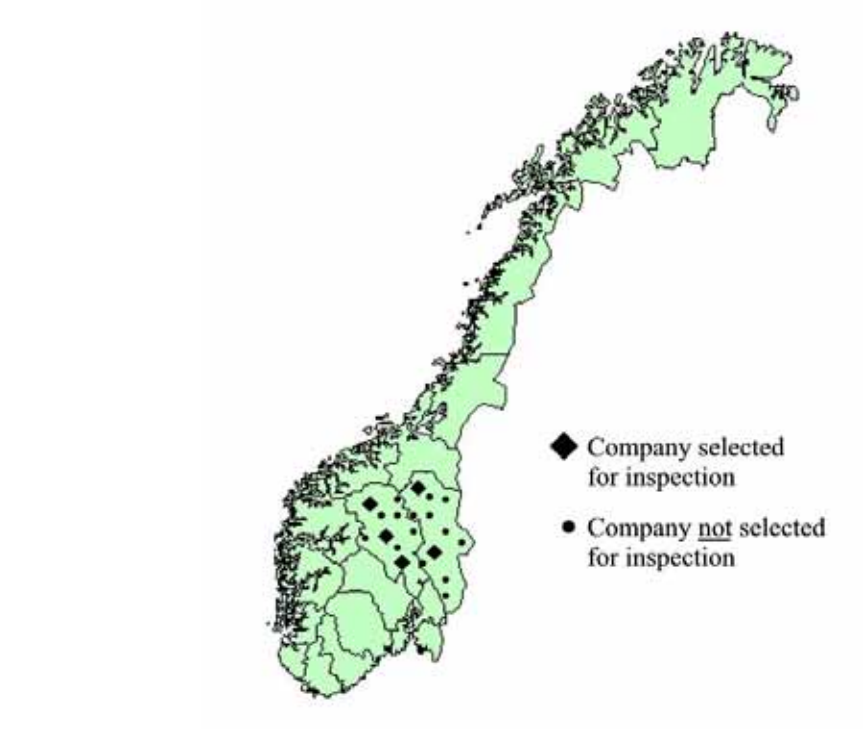

Fig. 1: Illustration of random regional inspection.

Inspection resources

All kinds of inspections require time for preparation, performance and complementary work. Most inspections require involvement of two or more persons. NRPA almost never perform unannounced inspections.

Consumption of time and manpower: The time spent on preparation for the three inspection methods described in this paper is roughly the same, and we are talking about several man-days. Before thematic inspections and random regional inspections a lot of the preparation time is spent on contacting companies, organising travels and overnight lodging, while most of the preparation time for system audits is spent on studying the operator's documentation.

Also the man-days spent during the performance period are roughly the same for the three inspection methods. Thematic and random regional inspections can in theory be performed by one person, but two or more persons are preferred. With more than one inspector there are several options; inspecting the same company or several companies at the same time. Performance of system audits usually requires 3-5 persons, but the complementary work is roughly done in half the time compared to the two other inspections methods. Complementary work for system audits takes about one or two man-days, and twice the time for thematic and random regional inspections.

Costs: The lodging costs are roughly the same for the three types of inspection described in this paper, but thematic and random regional inspections usually have higher travelling expenses because the inspections take place in different companies which may be separated by a long distance. In an extended country like Norway, it is both time- and cost-saving to concentrate thematic and random regional inspections within a limited geographical area.

Inspection experience

Thematic inspections are well suited for large as well as small companies, preferably having a specific type of radiation equipment or dealing with a specific radiation protection problem (e.g. industrial radiation gauges, industrial radiography, irradiators or interventional radiological procedures). This type of inspection is not so time-consuming for the individual company, and result in extra focus on the radiation protection system for a period and gives a brief summary of the radiation protection status in the company. For the national authority thematic inspections gives a picture of the status for a certain type of radiation use, as well as increased competence in the specific field in question.

System audits are best suited for large companies, hospitals or universities having many radiation sources and radiation users. This type of inspections is very time-consuming for the individual company, but it gives a thorough picture of the radiation protection status in the company. A system audit can be a very useful tool for the company in the work of improving their radiation protection, but from the national authority point of view it only gives the opportunity to inspect one company for the same cost and time consumption as for many companies by use of other inspection methods.

Random regional inspection has been used with success by other regulatory authorities in Norway, and NRPA will test this method on users of industrial radiation gauges during 2005. The method can be compared to thematic inspections, but with wider effects because also companies not inspected focus extra on radiation protection in a period and receive relevant information.

Cooperation between HSE regulatory authorities in Norway

Several regulatory authorities in Norway perform inspections in the health, security and environment (HSE) area. NRPA became a part of the group of HSE regulatory authorities from 1 January 2004. A Royal decree of 17 September 2003 appoint the Norwegian Labour Inspection Authority as coordinating regulatory authority for HSE inspections onshore in Norway. Correspondingly, the Petroleum Safety Authority Norway is, in a Royal decree of 19 December 2003, given the coordinating role for offshore HSE inspections. The group of HSE regulatory authorities will in 2005 establish a new public web site including information on the most important HSE requirements for companies in Norway (see pilot site in Norwegian at www.regelhjelp.no) and inspections can be coordinated by using their own internal inspection database.

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Radiation protection regulatory authority in Lithuania

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Abstract: The purpose of this presentation is to present the legislation system and radiation protection regulatory authority in Lithuania, its responsibilities and tasks, development, principles and priorities, experiences and etc.

Introduction

The state infrastructure of radiation protection in Lithuania is still being created in connections with recommendations laid out in the ICRP-60 Publication, IAEA recommendations and requirements of legislation of European Community. This includes laws and regulations, an efficient regulatory system and supporting services.

Lithuania in order to protect individuals, society and the environment from the harmful effect of ionizing radiation has taken the appropriate measures necessary to ensure that radioactive sources are safely managed and securely protected and practice with them are under control and supervision during and the end of their handling.

Legislation and statutory framework

Laws: Four main laws - the Law on Radiation Protection, the Law on Nuclear Energy, the Law on the Management of Radioactive Waste and the Law on Environmental protection - prescribe the provisions for regulations of safety and security of sources of ionizing radiation to ensure the radiation protection of general public and workers working with sources of ionizing radiation (hereinafter - workers).

The Law on Radiation Protection has established the legal basis of radioactive protection allowing to safeguard people and the environment from the harmful effects of ionising radiation.

The Law on Nuclear Energy ensures nuclear safety when nuclear energy is used to meet peaceful needs and prevents any illegal disposition of nuclear materials, including nuclear fuel and nuclear waste.

The Law on the Management of Radioactive Waste has established the legal grounds for the management of radioactive waste.

The Law on Environmental protection regulates some aspects of radiation protection of the public relating to the control of releases of radioactive materials into the environment.

The main responsibilities in the field of radiation protection are laid on Radiation Protection Centre (RPC). The law on Radiation Protection has given to the RPC the status of radiation protection authority in Lithuania and has empowered the RPC to undertake all regulatory activities in the field of radiation protection.

Regulations: A comprehensive system of radiation protection regulations includes:

Resolutions of the Government of Lithuania (regulations on licensing of practices with sources, on register of sources and exposure of workers, on dosimetric control in the case of radiological accident, on management of illegal sources).

Orders of the Minister of Health (various types of regulations, including regulatory procedures and rules, as well as, the Statute of the RPC).

Orders of the Minister of Environment (regulations on radioactive discharges and the establishment of clearance levels).

Orders of the Minister of Health and the Minister of Environment (regulations on environmental monitoring and the monitoring of building materials).

Hygiene standards (regulations on main radiation protection standards).

Orders of the RPC Director (detailed requirements and procedures related to different practices).

The key regulations in the implementing of the IAEA recommendations as well as the European Community directives and recommendations in the field of radiation protection are Hygiene Standards. They include both general and practice-specific standards. The general standards (Basic Safety Standard of Radiation Protection, Radiation Protection and Safety of Outside Workers, Requirements for Monitoring of Internal Exposure) are applicable to all practices. Practice specific-standards cover medical (X-ray, nuclear, radiotherapy), industrial and research uses, natural radiation and others.

Radiation protection authority in Lithuania

The RPC is the main organization which co-ordinates the activities of executive and other bodies of public administration and local government in the field of radiation protection, exercising state supervision and control for radiation protection, monitoring and expert examination of public exposure. The RPC was established on January 1, 1997 by the Order of the Minister of Health. It is a self-sufficient budget funded institution, which acts according to the Constitution of the Lithuanian Republic, laws and other legal documents.

Implementing the main goal - to protect general public and workers against the harmful impact of ionizing radiation - the RPC carries out following functions:

- It drafts laws and other legal acts on public and occupational radiation protection.

- It keeps the State Register of Sources of Ionizing Radiation and Occupational Exposure.

- In accordance with the procedure established by the Government, it issues, reregisters, suspends, renews or revokes the licenses for conducting practices with sources of ionizing radiation.

- It carries out the radiation protection state supervision and control.

- It organizes and conducts monitoring of contamination by radionuclides of air, drinking water, foodstuffs and their raw materials, building materials and their products as well as other objects which may result in the exposure of humans.

- It carries out study of radiological accidents, forecasts their consequences and makes proposals for their prevention and containment.

- It prepares, within the limits of its competence, reviews of radiation protection and makes proposals relating with regard to radiation protection to the executive bodies of public administration, control institutions and local government of all levels.

- It provides information to the public.

Regulatory activities of the radiation protection authority

Notification and inventory of sources: Radiation protection of humans and environment may be assured if the state maintain data about sources which are used in the territory of the state. For this purpose the State Register of Sources of Ionizing Radiation and Exposure of Workers has been established and its statute approved by the Decree of the Government. According this Decree the RPC is responsible for maintaining the Register. The aim of the Register is to collect, compile, systematize, store and provide the data on sources, which are used in Lithuania, and workers according to the order established by legal acts.

The requirement to maintain the register of sources is determined by the order of the Minister of Health. Current regulations require users to carry out an annual inventory of the sources and report this to the RPC every 12 months. The owner of the source, having acquired, sold, installed, decommissioned or disposed of it, shall fill in the appropriate report form and present it to the RPC within 10 working days.

Licensing of practices with sources: According to the Government decision, the manufacture, usage, storage, installation, maintenance, repairs, processing, transportation, sale of sources and management of radioactive waste without appropriate license is forbidden.

The cases exempt from licensing requirements are the following - exempt sources are used in practice, generators of ionizing radiation are transported or stored and radioactive materials of the first category of danger marked with white label according to the European Agreement concerning the International Carriage of Dangerous Goods by Road are transported.

The following types of licenses are determined - for manufacture, usage, storage, maintenance, repairs, processing of sources of ionizing radiation and management (collection, segregation, processing, storage, reprocessing, transportation, disposal) of radioactive waste, for installation of sources, for sale of sources and for transportation of sources and radioactive waste.

The license is granted to the legal persons after they send the written application and all necessary documents what prove that they understand the main principles of radiation protection (justification of practice, optimization of radiation protection and limitation of doses) and are ready to follow the requirements of legal acts that ensure radiation protection of people and workers.

Inspections and enforcement: Frequencies of inspections based on the risk categories of source, the planning of inspections, types of inspections and the responsibilities of inspectors are defined by the order of the Minister of Health. Inspectors are also in possession of a manual on inspections, which contains written procedures and protocols.

On the basis of results of performed inspection an inspection protocol is filled in and time period for eliminating violations of the requirements of radiation protection and source safety is determined. If it is not done within prescribed time, the RPC has the right to suspend or repeal the license or impose a fine.

Information dissemination: The information system has been established to inform the public and workers about the radiation safety aspects of regulated practices, intervention situations and the regulatory process. A number of guidance documents have been made available to licensees. Reports on events and fundamental questions in radiation protection are routinely made available to the public using all appropriate media and RPC WebPages.

Improving of regulatory functions

International co-operation is one of the most important tools in creation of well functioning national radiation protection authority and improving of RPC functions. Intensive work in development of radiation protection infrastructure and many efforts in strengthening of the RPC has been done in national technical co-operation projects with the IAEA, in the framework of Phare program “Radiation protection” and the continuous help from Sweden and Finland experts. During these activities the radiation protection legislation and radiation protection infrastructure, radiation protection in management of radioactive waste, emergency preparedness system has been improved, creation of systems of radiation protection training and public information and creation of quality system of RPC has been begun.

Conclusions

The result of creation of radiation protection infrastructure and regulatory authority in Lithuania is the modern radiation protection authority responsible for radiation protection of population, workers and environment and implementation of EC radiation protection legislation. Further steps in the activities of RPC are the development of the quality management system of regulatory functions and achievement to have opportunity to share experience with other East Europe countries.

Radiation protection programmes for transport of radioactive material

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Abstract: One of the new requirements put forth in the transport regulations is that a radiation protection programme shall be established for the transport of radioactive material, and that such programmes shall be available, on request, for inspection by the relevant competent authority. SSI has developed guidelines for such programmes and by following these guidelines it is believed that the doses received by workers and members of the public will be kept as low as reasonably achievable social and economic factors being taken into account (ALARA) without introducing unnecessary regulatory or economic burden to transport organizations.

Background

Since 1961, the IAEA has been publishing recommendations covering the safe transport of radioactive material. These recommendations undergo periodic revisions including those in 1973, 1985 and 1996. The 1996 Edition was the result of a ten-year review process. It was published as "Regulations for the Safe Transport of Radioactive Material", 1996 Edition, Safety Standard Series No. ST-1 [1], and was reissued with minor corrections in 2000 as TS-R-1 (ST-1, Revised) [2] and again as 1996 Edition (As Amended 2003), Safety Standard Series No. TS-R-1 [3].

One major topic in that revision process with relevance to radiation protection is the incorporation of the provisions included in "International Basic Safety Standards for Protection Against Ionizing Radiation and for the Safety of Radiation Sources", IAEAs Safety Series No.115 [4], which reflects the recommendation in ICRP Publication 60 [5]. The BSS emphasize the importance of the three basic radiation protection principles

- Justification of a practice; no practice shall be adopted unless it produces a net benefit
- Limitation of dose and risk to individuals; exposure to individuals should be subject to dose and risk limits
- Optimization of radiation protection and safety; all exposures shall be kept as low as practicable, economic and social factors being taken into account.

Optimization of radiation protection has since a long time been the driving force behind the development of protection in most practices involving ionizing radiation and it is expected that the same will now be the case in transport of radioactive material.

One of the major new requirements in the 1996 Edition of the Transport Regulations [1] is that those involved in the transport of radioactive material must establish a Radiation Protection Programme (RPP). An RPP should contain systematic arrangements that are

aimed at providing adequate consideration of radiation protection measures. The RPPs shall, on request, be available for inspection by the competent authority.

The IAEA Transport Regulations [1,2,3] has been transposed to separate but consistent international and national regulations for transport by road, rail, sea and air, which entered into force around 2001. For example for transport by road in Sweden by Statens Räddningsverks föreskrifter om transport av farligt gods på väg och i terräng [6].

IAEA has produced guidelines on the content and applications of an RPP, which will soon be publish as a Safety Guide [7]. Based on that the Swedish Radiation Protection Authority has published a report with guidance relevant for Swedish users [8]. This paper is a short summary of the SSI report, which can be ordered from SSI or downloaded from www.ssi.se.

Summary of Guidance

Optimization of radiation protection and safety, i.e. all exposures shall be kept as low as reasonably achievable, economic and social factors being taken into account (ALARA principle), should be the driving force behind the RPP. Sometimes it may be relevant to use mathematical methods but mostly common sense and good practise is good enough as stated by ICRP [9]:

"Much can be achieved in optimization of protection, particularly in everyday operational control, through the use of professional judgement by suitably qualified, experienced and competent persons. The following are suggested to help judge if an action is reasonable:

(a) Common Sense; this reflects experience, knowledge and the exercise of professional judgement. For example, a very low cost yet practical change that reduces dose probably should be done even if doses are already low.

(b) Good Practice; this compares what has been or is expected to be achieved with what has been achieved for similar or related facilities or practices. Care must be taken to ensure that reasonableness is maintained and that unwarranted expenditures do not become the norm. "

RPPs are intended to provide and document in a systematic way the controls applied by transport organizations to satisfy the basic radiation protection principles. The RPP should be a help to

- provide for adequate consideration of radiation protection measures,
- ensure that the system of radiological protection is adequately applied,
- enhance the safety culture, and
- provide practical measures to meet these objectives.

An RPP may consist of one or several documents and can be a "stand alone document" or be part of the operator's Quality Management System. The RPP should be a living document which should be revised regularly taking into account changes in regulations, transport conditions, contents, transport volumes, equipments used, modes of transport and experience gained during previous shipments.

It is recommended that the extent of control measures included in the RPP should be related to the magnitude and likelihood of radiation exposures, i.e. the control measures are expected to be commensurate with the level of hazards arising from transport of radioac-

tive material - “graded approach”. The first step is therefore to make a prior assessment of the situation including a description of the type and volume of the radioactive material, the magnitude and likelihood of radiation exposure expected to arise from these shipments, number of workers involved, duration of operations and distances from the radioactive packages to the workers and to members of the public.

Operations involving only a limited number of packages containing small quantities of radioactive material may require only a short RPP while complicated operations involving diverse materials and packages and handled in the public domain need a more significant RPP covering all aspects involving routine, normal and accident conditions of transport.

A radiation protection programme should at least cover:

(a) Scope and responsibilities for the implementation of the programme.

(b) Dose assessment and optimization.

(c) Surface contamination assessment.

(d) Segregation and other protective measures.

(e) Emergency response arrangements

(f) Training and information

(g) Management systems for quality assurance.

It should be noted that low occupational dose does not mean an RPP is not required. For instance transport of large quantities of radioactive material with high activities in heavily shielded packages generally gives rise to low doses but still requires consideration of emergency response and training.

The following table with a summary of the basic elements of an RPP can be found both in the IAEA Safety Guide [7] and in the SSI Report [8]

Table. RPP Elements versus occupational dose.

RPP element	**Occupational dose**		
	Not more than 1 mSv in a year	More than 1 mSv in a year but not more than 6 mSv in a year	More than 6 mSv in a year
Scope	Yes		
Roles/Responsibilities	Yes		
Dose Assessment	No monitoring required	Workplace or individual monitoring	Individual monitoring mandatory
Dose Limits / Constraints/Optimization	Yes, but basic optimization	Yes	

Surface contamination	Must be considered
Segregation and other protective measures	Only applicable to II-YELLOW, III-YELLOW, III-YELLOW under exclusive use (and packages containing fissile material)
Emergency Response	Yes
Training	Yes
Quality Assurance	Yes

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[1] INTERNATIONAL ATOMIC ENERGY AGENCY, Regulations for the Safe Transport of Radioactive Material, 1996 Edition, Safety Standards Series No. ST-1, IAEA, Vienna (1996).

[2] INTERNATIONAL ATOMIC ENERGY AGENCY, Regulations for the Safe Transport of Radioactive Material, Safety Standards Series No.TS-R-1 (ST-1, revised), IAEA, Vienna (2000).

[3] INTERNATIONAL ATOMIC ENERGY AGENCY, Regulations for the Safe Transport of Radioactive Material, Safety Standards Series No.TS-R-1, 1996 Edition (As Amended 2003), IAEA, Vienna (2004).

[4] INTERNATIONAL ATOMIC ENERGY AGENCY, International Basic Safety Standards for Protection against Ionizing Radiation and for the Safety of Radiation Sources, Safety Series No. 115, IAEA, Vienna (1996).

[5] INTERNATIONAL COMMISSION ON RADIOLOGICAL PROTECTION, ICRP Publication 60, 1990 recommendations of the International Commission on Radiological Protection, (1990).

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[7] INTERNATIONAL ATOMIC ENERGY AGENCY, Safety Standards, Radiation Protection Programmes for Transport of Radioactive Material DS 377, Safety Guide (to be published).

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Simplifying licence handling through ”register extraction”

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SIMPLIFYING LICENCE HANDLING THROUGH ”REGISTER EXTRACTION”

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Within the division of research and industry at the Swedish Radiation Protection Authority (SSI), the licence certificates for activities with ionizing radiation sources has been simplified through introducing “register extraction”. This process is simplified through our new system with a single licence and a register extraction.

FIRST TIME APPLICATION

- for every new licence holder with activity involving ionizing radiation
- information about the licence holder
- signed by the person with procuration

NEW REGISTRATION

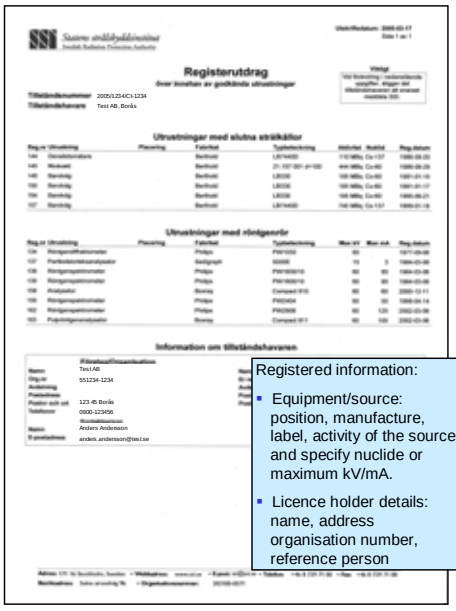
- new registration is sent in for every new equipment/source

UNREGISTRATION

- unregistration is sent in for every unregistration of equipment/source

The registration forms can be found through our website www.ssi.se.

REGISTER EXTRACTION



Registered information:

- Equipment/source: position, manufacture, label, activity of the source and specify nuclide or maximum kV/mA.
- Licence holder details: name, address organisation number, reference person

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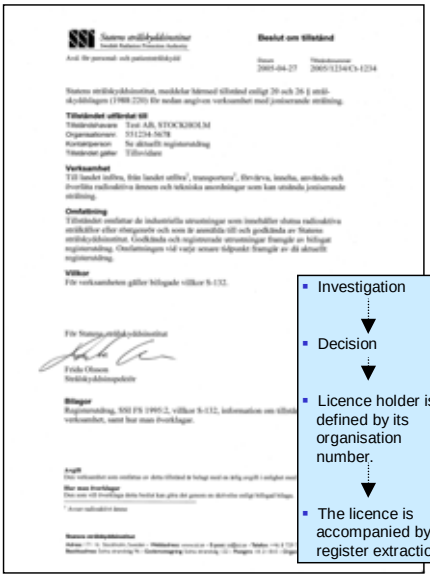
LICENCE



- Investigation
- Decision
- Licence holder is defined by its organisation number.
- The licence is accompanied by a register extraction.

This way of handling licences with register extractions simplifies our work:

- one licence holder ⇒ one licence
- licence holder receives an updated register extraction which keep both our and their information updated

Introducing HASS

- each radioactive source and the equipment has to be registered

SUMMARY

- Each licence is complemented with a register extraction
- Simplifies the administrative work when a company with a licence wants to register/unregister a source/equipment.
- The licence holder becomes aware of what is registered at SSI which helps them keep our information updated.

Expert Panel Elicitation

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Abstract: Scientists are now frequently in a situation where data cannot be easily assessed, since they may have conflicting or uncertain sources. While expert judgment reflects private choices, it is possible both reduce the personal aspect as well as in crease confidence in the judgments by using formal protocols for choice and elicitation of experts. A full-scale elicitation made on seismicity following glaciation, now in its late phase and presented here in a preliminary form, illustrates the value of the technique and some essential issues in connection with the decision to launch such a project. The results show an unusual low variation between the experts.

The technique of expert panel elicitation, EPE

Examples of fields that have contributed to probability elicitation are decision analysis, psychology, risk analysis, Bayesian statistics, mathematics and philosophy. In the safety assessment of the final disposal of radioactive waste, expert judgment in many different forms will undoubtedly play a significant role. Quantification of subjective probabilities is employed in a number of circumstances.

These include, particularly regarding political aspects:

- when there is likely to be public scrutiny of the uncertainties
- in situations requiring impartial judgments
- in cases where there are potential legal action,

combined with scientific problems occurring:

- when data exist only from analogue situations
- when the uncertainties are significant relative to the demonstration of compliance

Several, if not all, of these criteria are relevant to the final disposal of radioactive waste, both the bullets about public scrutiny and need for impartial judgment and need for handling uncertainties.

Formation of a research project

A common research project was suggested by SSI at a meeting in November 2004 based on the assumption that both operator and regulators may have an interest in the method of expert panel elicitation,. SSI had invited the operator, the Swedish Nuclear Fuel and Waste Management Company, SKB, the Swedish Nuclear Power Inspectorate, SKI, and representatives from the two municipalities involved in SKB's ongoing site investigation (2005), Östhammar and Oskarshamn. At the meeting Prof. Stephen Hora from the Uni-

versity of Hawaii at Hilo was invited as an expert on the method. The project is thus not set up as a part of the ongoing operator-regulator license activities but as a pure research project of interest to all parties. The practical initiatives have been made by SSI and SKI, mainly through Mikael Jensen and Eva Simic. SKB's representative Raymond Munier, took part in an observing capacity, and SKB also offered to place its databases at the project's disposal, to meet the need for such data, as deemed appropriate by the experts. The project is financed jointly by the authorities SSI and SKI and the operator SKB.

Elicitation issues

The quantity or quantities for elicitation were discussed at the first preparatory meeting in 24 February 2005. The municipalities' representatives were in favour of seismicity issues but several possibilities were discussed, e.g. the future fate of the Baltic Sea shoreline. Finally, two questions on seismicity following glaciation were defined, related to seismicity frequency after a glaciation, and earthquake consequences for a repository.

The procedure for nomination and selection

A letter inviting 30 organisations to nominate was sent out. These the same group of organisations or stakeholders that have shown an interest in the Swedish Waste Program and e.g. are usually invited to review SKB:s research plan, including government authorities, universities and environmental groups. While this was an open and transparent method, it became obvious after a while that it did not yield a sufficient number of experts. Within the project reference group a number of additional experts were therefore also nominated. In the end of the process, 16 experts remained. Four additional experts were chosen to form a selection group, who selected 5 experts among the 16 nominees. The selection group was made up by Jimmy Stigh, University of Gothenburg, Roland Roberts, Uppsala University, Ove Stephansson, the Swedish Royal Technical University and Giogio Ranalli, Carleton University (Canada). The experts chosen were i) John Adam, NRC, Geological Survey of Canada, ii) Hilmar Bungum, University of Oslo, iii) James Dieterich, University of California, Riverside, USA, iv) Kurt Lambeck, University of Canberra, Australia and Björn Lund, University of Uppsala, Sweden.

The expert's assignments

The experts were asked to attend a two-day meeting where the elicitation methods were explained, including the technique of expressing uncertainty as subjective probabilities, which was tested by a calibration exercise using a quiz of questions outside the experts' fields. Also the elicitation questions were discussed in detail. The first question was simplified in different ways, but it was apparent that more work was needed for the first issue than anticipated. One of the experts, Kurt Lambert, agreed to calculate a stress field for a scenario similar to the Weichselian glaciation and report his findings back to the group.

The expert group's first meeting 17-18 May

It became clear from the discussion between the experts at the first expert meeting that it would be difficult to cover both questions based on the time deemed necessary for ques-

tion 1 alone. Given the financial restrictions of consultancy between the two meetings, it was therefore necessary to focus on one of the two questions, and question 1 was chosen. During the first expert meeting the original question 1 was further negotiated to have the following formulation:

> *What will be the frequency of moment magnitude 6.0 or greater earthquakes per unit area (e.g. per 100 sq. km) in the middle and south of Sweden (Forsmark and Oskarshamn) during a glacial cycle (appr 100 000 a) assuming conditions similar to the Weichel glaciation? Give an uncertainty distribution for this quantity for each area.*

Some additional conditions were also assumed, as well as some specified relevant reading material that was made available to all experts. During the problem definition meeting it was noted that none of the experts seemed to have a spontaneous feeling for the result to come. All experts relied on calculations to be performed later.

The elicitation

At the second meeting 20-21 of June the experts met and gave a short presentation of their work. After the presentations, the 5 experts' probabilities were elicitated separately by the elicitator Stephen Hora in the presence of one from the project group, Mikael Jensen. Kurt Lambeck gave data in a form that needs transformation in a way that makes it problematic to transform his data to the format, and his contribution to the elicitation is not presented here.

The elicitation probabilities for the two areas Oskarshamn and Forsmark were collected and transferred to cumulative probabilities in Table 1 and 2.

Table 1. Estimated Cumulative probability of earthquakes in a 100 km^2 area around Oskarshamn

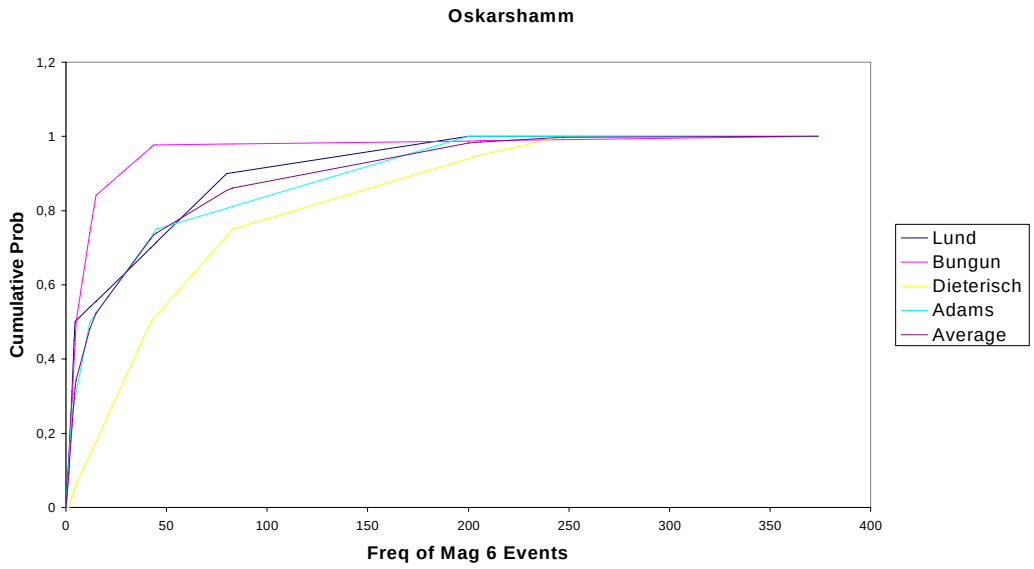
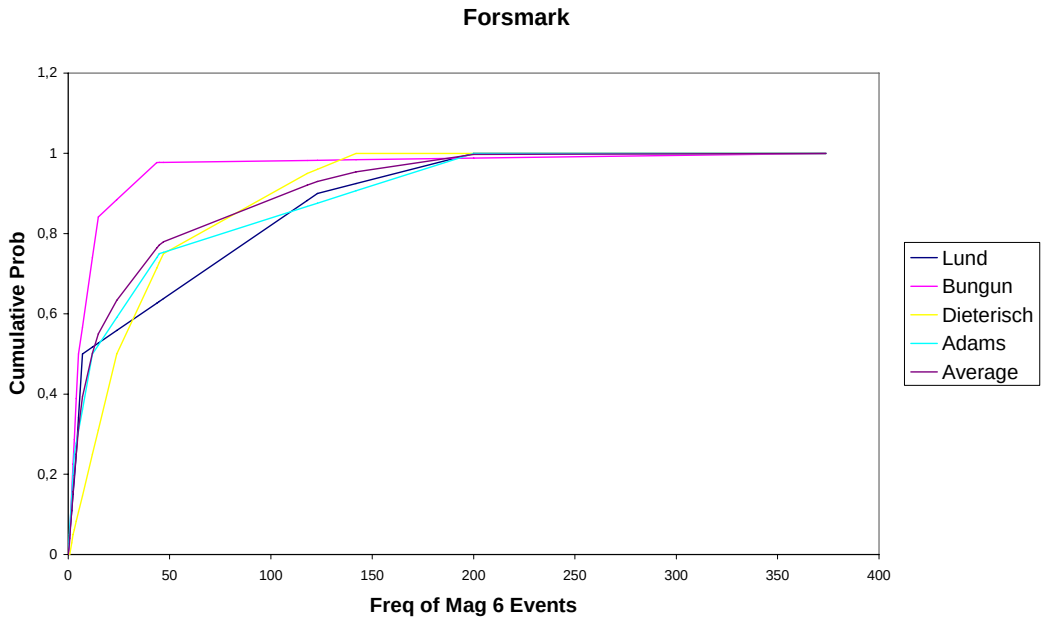


Table 2. Estimated Cumulative probability of earthquakes in a 100 km^2 area around Forsmark



Results

Only preliminary results are presented as the evaluation is still going on. However, it can be seen that all results are below 200 earthquakes per 100 km^2 for a complete glaciation cycle and for the Forsmark area, the 50 cumulative probability roughly lies, with a surprisingly low spread, between 10 and 30. For Oskarshamn, the spread was only modestly greater, from about 10 to 40. There was not a great difference between the two sites.

Discussion

One component that might lead to expectation of uniformity was that Kurt Lambeck calculated stress load over the full glacial cycle and made that available to the group. However, John Adams took a considerably different approach, relying mainly on observational data from a Swedish paleogeologist, Nils-Axel Mörner, and yet the curve obtained from John Adams lies very close to the average of all experts in both figures.

It is also worth noting that it may be difficult to mimic the situation behind a regulatory motivated elicitation with a research elicitation such the one presented here. In a license application dialogue, an issue may be seen as unresolved or unsatisfactorily illustrated at a late stage, even after considerable effort directed to the issue from both regulator and operator. In an attempt to solve the problem of that still unresolved issue, an elicitation may be used. In this case there may be reasons to expect that the issue is much better defined by the nature of the process that lead to the situation at hand. In the present work the issues simply reflected a general interest at a meting between the project members.

In any case, it is obvious from our experience that it is worthwhile to direct considerable effort towards a strict problem formulation that includes an estimate of time needed for the elicitation.

SSI-rapporter 2005

SSI reports 2005

2005:01 Reports from SSI:s International Independent Expert Group on Electromagnetic Fields 2003 and 2004.
SSI's Independent Expert Group on Electromagnetic Fields 190 SEK

2005:02 (SKI 2005:02) International Peer Review of Swedish Nuclear Fuel and Waste Management Company's SR-Can interim report
Budhi Sagar, Lucy Bailey, David G Bennett, Michael Egan, Klaus-Jürgen Röhlig

2005:03 (SKI 2005:06) Granskning av SKB:s SR-Can interimsrapport:SKI:s och SSI:s bedömning av SKB:s uppdaterade metoder för säkerhetsanalys
Benny Sundström och Björn Dverstorp et. al.

2005:04 (SKI 2005:10) Concentrations of Uranium, Thorium and Potassium in Sweden
Bo Thunholm, Anders H. Lindén och Bosse Gustafsson 130 SEK

2005:05 (SKI 2005:32) Säkerhets- och strålskyddsläget vid de svenska kärnkraftverken 2004
SKI och SSI

2005:06 Percutan coronar intervention PCI – en strålskyddsutredning av verksamheten på landets sjukhus
Avdelningen för patient- och personalstrålskydd
Anja Almén, Torsten Cederlund och Britta Zaar 70 SEK

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Avdelningen för beredskap och miljöövervakning
Hans Möre och Lynn Marie Hubbard 80 SEK

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SKI och SSI
Anders Wörman, Björn Sjögren och Lars Marklund

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SSI Internationellt Utvecklingssamarbete, SIUS
Sten Grapengiesser och Torkel Bennerstedt 120 SEK

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Avdelningen för beredskap och miljöövervakning
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2005:15 Radiological Protection in Transition - Proceedings of the XIV Regular Meeting of the Nordic Society for Radiation Protection, NSFS - Rättvik, Sweden, 27-31 August 2005
Redaktörer: J. Valentin, T. Cederlund, P. Drake, I.E. Finne, A. Glansholm, A. Jaworska, W. Paile och T. Rahola 600 SEK

SSI report

STATENS STRÅLSKYDDSINSTITUT, SSI, är central tillsynsmyndighet på strålskyddsområdet. Myndighetens verksamhetsidé är att verka för ett gott strålskydd för människor och miljö nu och i framtiden.

SSI är ansvarig myndighet för det av riksdagen beslutade miljömålet *Säker strålmiljö*.

SSI sätter gränser för stråldoser till allmänheten och för dem som arbetar med strålning, utfärdar föreskrifter och kontrollerar att de efterlevs. Myndigheten inspekterar, informerar, utbildar och ger råd för att öka kunskaperna om strålning. SSI bedriver också egen forskning och stöder forskning vid universitet och högskolor.

SSI håller beredskap dygnet runt mot olyckor med strålning. En tidig varning om olyckor fås genom svenska och utländska mätstationer och genom internationella varnings- och informationssystem.

SSI medverkar i det internationella strålskyddssamarbetet och bidrar därigenom till förbättringar av strålskyddet i främst Baltikum och Ryssland.

Myndigheten har idag ca 110 anställda och är belägen i Stockholm.

THE SWEDISH RADIATION PROTECTION AUTHORITY, SSI, is the government reguatory authority for radiation protection. Its task is to secure good radiation protection for people and the environment both today and in the future.

The Swedish parliament has appointed SSI to be in charge of the implementation of its environmental quality objective *Säker strålmiljö* ("A Safe Radiation Environment").

SSI sets radiation dose limits for the public and for workers exposed to radiation and regulates many other matters dealing with radiation. Compliance with regulations is ensured through inspections.

SSI also provides information, education, advice, carries out its own research and administers external research projects.

SSI maintains an around-the-clock preparedness for radiation accidents. Early warning is provided by Swedish and foreign monitoring stations and by international alarm and information systems.

The Authority collaborates with many national and international radiation protection endeavours. It actively supports the on-going improvements of radiation protection in Estonia, Latvia, Lithuania, and Russia.

SSI has about 110 employees and is located in Stockholm.

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