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Nordic society for radiation protection - NSFS

Proceedings of the NSFS XV conference in Ålesund Norway, 26-30 of May 2008

Statens strålevern
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Abstract:
The report contains written versions of the presentations at the conference in Ålesund, arranged by The Nordic society for radiation protection 26–30 of May 2008. Included are full text versions, or abstracts, from all sessions, i.e: opening, medical, effect studies, natural radiation and radiation in nature, emergency preparedness, radioactive waste and nuclear safety and security.

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Østerås, 2008

Nordic radiation protection cooperation has long traditions, in respect to cooperation between regulators as well as scientists. The Nordic society for radiation protection (NSFS) is an associate of the International radiation protection association (IRPA), and the members of NSFS are professionals representing all different areas related to radiation protection.

Ordinary meetings of NSFS are arranged every 3rd year, representing an arena for young scientists to present their work and for regulators to discuss common issues. The XV ordinary meeting was arranged in Ålesund 26 – 30 of May 2008 with approximately 100 participants, in a surrounding of mountains and fjords.

The receiver of the Bo Lindell award, Sten Carlsson, gave an introductory lecture on the topic "Education and training- the basis of radiation protection for medical exposure". Separate sessions were held on medical radiation, effect studies, natural radiation and radiation in nature, emergency preparedness, radioactive waste and nuclear safety and security.

In all there were 40 oral presentation and 10 posters, in addition to two invited lectures on "The 2007 recommendations of the international commission on radiological protection" and "Thorium as energy resource – pros and cons".

As president of NSFS I wish to thank all oral and poster presenters, and other participants for taking active part in the discussions. I also wish to thank the Norwegian radiation protection authority and the Institute for energy technology for sponsoring the arrangement. Likewise I wish to thank the participating vendors displaying the latest developments in measuring equipment for sponsoring the meeting through their participating fee.

Østerås September 29th 2008

Tor Wøhni

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

Innhold/Contents

1 Opening session

Bo Lindell Lecture. Education and training - the basis of radiation protection for medical exposure

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It is a great honour for me to have received this invitation to present the Bo Lindell lecture for 2008 and I sincerely thank the Nordic Society for Radiation Protection.

This lecture is based on my own experiences from work as a hospital physicist in an ordinary sized Swedish hospital and gained from experiences of several international training courses organized by IAEA. The lecture deals only with medical exposure in nuclear medicine and the diagnostic and interventional use of x-rays.

Introduction

Medical exposure is the largest man-made source of radiation exposures. It accounts for more than 95% of the exposure from those sources and is increasing due to an increasing number of people undergoing x-ray examinations. Also, many x-ray examinations are now performed using CT instead of conventional equipment, which results in higher exposure of the patient. Finally, the increased use of fluoroscopically guided interventions has also contributed significantly to the increasing collective effective dose.

Although the possibility to injure the patient was known early, the main interest at the beginning of the medical use of ionizing radiation was concentrated on protection of the staff. It was not until the 1950:s that medical exposure was recognized as a major source of human exposures. UNSCEAR and ICRP collected data on patient doses and ICRP gave basic recommendations on this issue (1, 2, 3). Today much of the work by ICRP in the area of medical exposure is dealing with the application of their basic recommendations into the daily practice of radiology (4, 5, 6, 7, 8), supplemented with some training material (9).

The main objectives of radiation protection in diagnostic medical exposures are 1) to reduce the exposure in diagnostic imaging and 2) avoid radiation injuries in interventional procedures.

A high standard of radiation protection in a hospital, including protection of the patient, is generally based on a radiation protection organization with well defined responsibilities and a program for quality assurance. It should all be documented. However, radiation protection in the daily work of a department is basically a result of the attitudes of the people involved in the work and their willingness to adopt a safety culture regardless of any documents. Those attitudes can only be shaped through education and training.

All individuals involved in the occupational use of x-ray equipment and radioactive materials, require basic and continuing education and training in radiation protection and safety. Such training should be an integral part of their general professional education, which should include a basic course in radiation physics and radiation protection, together with a more detailed knowledge appropriate to their own professional sphere. All personnel in the practice should also have a thorough knowledge of the radiation protection program in their facility and should be well prepared to handle emergency situations.

The state of the practice around the world varies markedly. Many times the number of professionals contributing to radiological health care is very limited. In some circumstances a nurse or radiographer may be the only person available. However, training is still essential, perhaps even more in such cases, in order to meet the medical goals of managing the benefits against the risks of the procedures.

Personnell influencing on medical exposure

There are many categories of personnel in the hospital that are involved in medical exposure and thus having responsibilities also in radiation protection:

1. The *referring physician* who justifies a certain type of examination in order to reach a correct diagnosis and treatment.
2. The *specialist in radiology or nuclear medicine* who is responsible for the methods used in the department. He/she is also responsible for the use of optimized methods but should also take part in the justification by selecting the correct method from the request and in cooperation with the referring physician.
3. *Other specialists* e.g. cardiologists, orthopaedics etc. are responsible for the correct and optimized use of the equipment in their work with fluoroscopy.
4. The *radiographer/technologists* are the ones who generally perform the examinations. He/she is responsible for the correct use of the equipment and for following the methods approved by the department. He/she should correctly identify the patient and give the necessary information about the examination and be able to modify the exposure in accordance with any special needs for a single patient.
5. The *medical* or *hospital physicist* is responsible for optimization of the methods together with the specialist. He/she is also responsible for setting up the local radiation protection programme including dose measurements and education and training. He/she is generally also responsible for quality control and testing of the equipment. The physicist has the key position and should be the expert in radiation protection.
6. The *maintenance engineer* is responsible for the correct function of the equipment and is usually taking part in the quality control programme together with the hospital physicist.
7. The *license holder* as represented by the *hospital manager* and the *head of the department* has an overall responsibility for radiation protection. He/she has to establish the radiation protection organization, make sure that economical resources are available and to ensure that all workers receive adequate training in radiation protection and safety.

The *patient* himself will have responsibility in the protection of members of the family in case of treatments with radiopharmaceuticals or permanent implants with radioactive sources. The patient should be able and motivated to follow any instructions given regarding this issue. Patients cannot be expected to be well educated in matters of radiation risk and must be objectively guided in the benefit versus risk decision of their physician.

Basic education in radiation protection

The need for basic education in radiation protection varies among the personnel depending on their levels of responsibilities. General guidelines can be found in a publication from the European Commission (10). The highest educational level is required for the *medical physicist* and in developed countries this is generally achieved already in their basic education. To become a certified or accredited medical physicist, studies at the university level for 4-6 years is generally required which also include a comprehensive course in radiation protection (11). However there are also differences in the education among the different countries also within the European Community and organizations such as EFOMP (European Federation of Organisations for Medical Physics) is working towards a harmonization (12). Many countries throughout the world have no education of medical physicists at all and this will interfere with the standard of radiation protection in the hospitals. In such countries the only solution is to get the education abroad. IOMP (The International Organization for Medical Physics) provides a global listing of graduate education programs in medical physics as an aid to students (13). It is remarkable that only some institutes in Finland and none from the other Nordic countries are on that list. According to my personal opinion we should be more open-minded in accepting students from other countries. In some sense it is our responsibility, bearing in mind that the basic training of medical physicists in the Nordic countries is of a high standard.

IAEA offers postgraduate educational courses (PGEC) in radiation protection and the safety of radiation sources in five of its six official languages. The aim of the course is to provide an initial basic professional training in radiation protection and safety for young professionals who will later become trainers (14).

It is important that medical physics and radiation protection is recognized as an independent specialty which comprises highly educated people and should therefore have the same status as other specialties in the hospital.

The *medical doctor* in general will have no basic training in radiation protection. This is true all over the world. At some medical schools there might be some information on medical physics including radiation protection. Bearing in mind that any medical doctor can justify an x-ray examination or nuclear medicine examination through a request, there ought to be some education in radiation protection.

The *specialist in radiology or nuclear medicine* should have a comprehensive course in radiation protection. In theory this is recognized in most countries of the world. However, in reality it is possible to become a specialist in radiology or nuclear medicine without proper knowledge in radiation protection in both developed and developing countries. The reasons are either that the course is only an option among other courses or no courses exist due to lack of competent teachers and institutes. However, it is obvious that a course in radiation protection should be an obligation for a specialist in radiology or nuclear medicine.

Guidelines for training in general radiology, issued by the former European Association of Radiology, recommend that a formal course in physical and biological aspects of diagnostic imaging should be an essential requirement of the Core of Knowledge for general radiology (15). The syllabus presented includes those elements of physics and radiobiology that are relevant to the work of a radiologist.

AAPM (American Association of Physicists in Medicine) has proposed a curriculum that describes the core physics knowledge related to medical imaging that a radiologist should know when graduating from an accredited radiology residency programme. It is very comprehensive although the depth and order of presentation is left to the institution (16).

Several other organizations and societies have recognized the problem and made recommendations. The ambition is high but the reality is different and as mentioned earlier it is possible to find specialists with very limited knowledge in radiation physics and radiation protection.

Other specialists using fluoroscopy in their work will normally have no courses in radiation protection in their basic training programme. However, there is a need to integrate a basic course.

The basic education of *radiographers and technologists* will vary not only between countries of the world but also between countries in Europe and even within a country. In general the basic programme will have a length of 3-4 years but in some countries it will be considerably shorter. You can also find smaller hospitals with persons handling x-ray equipment with no education at all (17). Generally radiation protection is part of a course in physics and technology. For instance at one of the Swedish universities such a course will have a total duration of about 10 weeks with a relatively comprehensive number of lectures and discussions in radiation protection (18).

The Society of Nuclear Medicine has published a document on the performance and responsibility guidelines for nuclear medicine technologists, which also specifies their duties in radiation safety (19). Such a document defines the subjects that should be included in the basic training.

Maintenance engineers will generally have no education in radiation protection in their basic training. The same applies for persons with only administrative jobs, such as the *hospital manager* or equivalent.

Supplemental and continuing education and training

A well trained and motivated staff is fundamental to an effective radiation protection organization and the safety of other personnel and patients. However, it is obvious from what is said above that there are deficits in the basic education in radiation protection among many of the personnel influencing on medical exposure. This deficit has been recognized by several national and international organizations and professional societies who now offer courses in radiation protection as supplement to the basic training.

For the referring physician there will be a limited number of courses aimed to increase the knowledge in radiation protection. However, in the so called “patient directive” from the European Commission the following statement is made: “Member States shall encourage the introduction of a course on radiation protection in the basic curriculum of medical and dental schools”, so the situation will hopefully become better in the future (20). These ambitions will perhaps be adopted also in other countries. The European Commission has published a guideline aimed for the referral physician, where radiation protection is also taken into consideration (21). Some general information on radiological protection can be found in a supporting guidance from ICRP (22). For the hospital manager or equivalent training could include the basic principles of radiation protection, responsibilities relating to radiation risk management, the relevant legislation and regulations, the concept and principles of safety culture and the principal elements of a radiation protection programme. Such training should be organized on a regular basis and preferably by the Regular Authority, which will guarantee a high status of the course and recognition of the importance of radiation protection.

Many international organizations and societies arrange training courses on a regular basis to fill up the lack of knowledge in radiation protection for radiology specialists especially in developing countries but also for specialists working in the field of interventional radiology. For instance the IAEA has recently initiated training of cardiologists in radiation protection (23). The IAEA also has an extensive programme of training courses targeting education in radiation protection in general and in diagnostic imaging and nuclear medicine, ranging from the post-graduate courses to national and international

short-term courses (1-2 weeks). The courses are based on approved syllabus and training material (24, 25)

In many countries, radiation protection legislation requires continuing professional development (CPD) after the successful completion of initial training. This is to ensure that all individuals concerned with the use of radioactive materials have up-to-date knowledge. In some places special training courses may have to be organized for individuals who are already professionally qualified, but not trained in radiation protection. CPD should be an obligation for each individual working in the field of medical exposure in order to maintain the highest possible professional standards throughout their professional career. The types of events that might be part of a CPD system include national and international training courses, workshops, refreshing courses, congresses etc. One important aspect on training events in a CPD-system is that they should be accredited or in another way recognized by the Regulatory Authority or by a well established professional organization. Such accreditation or recognition should guarantee that certain standards are fulfilled on training facilities, teaching staff, content, material and methods for training, examination procedures and training records. The employer should provide the necessary resources for continuing professional development in radiation protection for all categories of staff working with ionizing radiation.

Besides regular training courses, a lot of material for continuing education and information can be found on the internet. The advantage is that it can virtually be reached by everyone working in the field of medical exposure. One example is the IAEA web-page on Radiological Protection of Patients (26). It contains sections on radiology, radiotherapy, nuclear medicine, interventional radiology and cardiology as well as other specialties such as PET/CT, dentistry and bone densitometry. The web-page also includes links to relevant literature and to the approved training material.

One example of this type of training material comes from the European Commission and is aimed for interventional cardiologists (27). It includes an evaluation consisting of 60 multiple choice questions. To pass the examination 75% should be answered correctly.

The Alliance for Radiation Safety in Paediatric Imaging consists of several professional organizations in the US. They have created a project named “Image Gently” as a guide to improve radiation protection in paediatric radiology. The web-page contains practical advices as well as some training material for different categories of personnel working in the field (28). The same subject is dealt with in a poster published by the ICRP in collaboration with the ISR (the International Society for Radiology) and Agfa (29).

The American College of Radiology (ACR) gives access to a variety of informative and instructional resources designed to assist radiologists in providing effective imaging and therapy while minimizing the risk during exposure to ionizing radiation (30). The US Food and Drug Administration (FDA) also provides valuable information for both professionals and patients (31).

Sprawls Educational Foundation has designed a training course in “Physical Principles of Medical Imaging”, which include presentations and a textbook on-line (32). The Swedish Society of Nuclear Medicine has also published a textbook on-line, which includes chapters on radiation biology, radiation risks and radiation protection (33).

Local education and training

For all personnel engaged in the use of equipment generating ionizing radiation it is essential that their basic education is supplemented with training on local equipment and local rules as well as the local organization of radiation protection. In many countries this is a legal requirement. For instance, in the Swedish regulations on x-ray diagnostics (34) this is expressed in the following paragraph:

"§ 8 All personnel in the practice shall have adequate education needed to perform the practice in a sound way from a radiation protection point of view.

Routines for such education shall be documented in writing in the quality manual. The document shall show which education elements different categories of personnel have to go through in order to be entitled to perform a certain work. For personnel working routinely with x-ray examinations of children, with screening or with large dose procedures like computed tomography or interventional radiology, particular high demands shall be required for education.

The personnel shall certify by signature that safety routines and other education elements have been gone through."

In the hospital such a regulation requires two types of education: 1) operation of the equipment and 2) radiation safety in general and the local organization and rules on radiation protection. The first one is often handled by the supplier in connection with the installation of new equipment. The second one should be organized by the medical physicist at two different levels depending on the duties and responsibilities of the personnel. One low level course, for example, 6-8 hours which also should include some practical demonstrations. Such a course is aimed at personnel with supporting duties for instance in the room for fluoroscopy either in the radiological department or in surgery. It is also useful for the specialists in orthopedics or general surgery who work with simple fluoroscopic equipment and normally use quite short fluoroscopic times. The main content of such a short course should be occupational protection. A higher level course of about 40 hours should also be organized on a regular basis. This course is aimed at the professionals with great influence on medical exposure such as the radiologist, interventional cardiologist and nuclear medicine specialist as well as radiographers working with CT and especially with pediatric patients. The course is also suitable for nuclear medicine technologists and the local maintenance engineers.

It is important that all personnel take part in the education organized locally. It is the responsibility of the head of the department to make sure that this is accomplished. Actually no one should be able to stay away unless he/she can present a certificate which shows that the necessary education has been received in another way.

Beside the courses, the medical physicist should also give general information on radiation protection to referral physicians, gynecologists and other professionals who may need it or ask for it.

Conclusions

Although medical exposure is different than occupational exposure and exposure of the general public because of the benefit to the patient it should still follow the general principles of justification and optimization. The most efficient way to increase the knowledge and awareness of radiation protection is education and training. To be most effective, education, training and continuous professional development in the radiological protection of patients must be an approach focused on all the contributors to the use of radiation in medicine, including specialists, referring physicians, radiological technologists, medical physicists, engineers and administrators. Training must target the specialty of the individual's practice and relate how their activities affect medical exposure.

Great efforts are made by national and international organizations as well as professional societies to increase the knowledge levels. However, there is still a lot to do especially in the basic training and as long as this situation last, a high standard of radiation protection in a hospital depends on the local resources for education and training. The specialist in radiation protection should be the medical physicist and therefore he/she, as a main duty, should have the responsibility to organize and carry out

education and training in the local hospital and with full support from the license holder. If all the personnel working in the field of medical use of ionizing radiation will take part in such an education, the standard of radiation protection will certainly increase in that hospital.

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ICRP Lecture. The 2007 Recommendations of the International Commission on Radiological Protection

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Introduction

The International Commission on Radiological Protection, ICRP, is an independent advisory organisation established in 1928 to advance the science of radiological protection. ICRP is the primary body in protection against ionising radiation. Its fundamental recommendations are revised at intervals of about 15 years, most recently in 2007. Development of these new fundamental recommendations took place over approximately ten years, and included two rounds of formal international public consultation in 2004 and 2006.

The main objective of the recommendations of ICRP is to provide an appropriate standard of protection for man and the environment without unduly limiting the beneficial practices giving rise to radiation exposure. This aim of providing an appropriate standard of protection, rather than the best possible standard regardless of costs and benefits, cannot be achieved on the basis of scientific concepts alone. Members of the Main Commission and its Committees supplement their scientific knowledge with value judgments about the relative importance of different kinds of risk and about the balancing of risks and benefits.

The biological basis

For deterministic effects (now preferably termed 'tissue reactions'), ICRP notes that 'in the absorbed dose range up to around 100 mGy (low LET or high LET) no tissues are judged to express clinically relevant functional impairment. This judgement applies to both single acute doses and to situations where these low doses are experienced in a protracted form as repeated annual exposures'.

The main focus now is however on 'stochastic effects', primarily cancer but also heritable disease. ICRP considers that for the purposes of radiological protection, it is scientifically plausible to assume that the incidence of cancer or hereditary disorders will rise in direct proportion to an increase in the equivalent dose in the relevant organs and tissues, below about 100 mSv.

ICRP also considered issues such as cellular adaptive responses, genomic instability and bystander signalling but notes that 'since the estimation of nominal cancer risk coefficients is based upon direct human epidemiological data, any contribution from these biological mechanisms would be included in that estimate'.

For other possibly radiation-associated health consequences (e.g., heart disease, stroke, digestive disorders, respiratory disease), ICRP concluded that the data are insufficient to allow for their inclusion in the estimation of detriment following low radiation doses (i.e., < 100 mSv).

The cancer risk estimates have not changed much since 1990. ICRP continues to consider that a dose and dose-rate effectiveness factor (DDREF) of 2 is appropriate for low doses and low dose rates.

For practical radiological protection purposes, the combined detriment from stochastic effects remained unchanged at around 5 % per Sv. The nominal estimates of total detriment are somewhat lower than in 1990 which is largely a reflection of the reduction in the estimated risk of serious heritable effects.

Dosimetric aspects

Radiation weighting factors (w_R), based on the RBE of the radiation of interest, are still used to derive the equivalent dose, but there has been some adjustment to the values used. The main changes are: the value for protons has been reduced from 5 to 2; charged pions have been introduced and also assigned a value of 2; the values for neutrons are now given only as a continuous function of energy (previously a step function was also recommended).

Revised values of tissue weighting factors (w_T) are given in the new recommendations, derived from the data on the risks of cancer induction and heritable disease. Four changes are noted. First, two more tissues have been included (brain and salivary glands). Second, the value assigned to gonads has been reduced from 0.20 to 0.05 reflecting the reduced significance of hereditary disease. Third, the w_T for breast has been increased from 0.05 to 0.12 in the light of recent epidemiological findings and the focus on cancer incidence in the detriment calculations. And fourth, the weighting of the so-called 'remainder tissues' has been modified so as to avoid earlier minor deviations from the desired additivity of effective doses.

The System of Radiological Protection

In the new Recommendations, ICRP retains the fundamental protection principles of justification, optimisation, and dose limits, and now clarifies how they apply to sources and to the individual.

However, ICRP moves from a process-based approach (practices or intervention) to an approach based on the exposure situation (planned, emergency, or existing). The concept of source-related dose and risk constraints is extended to all situations. Similar procedures are used for deciding on the extent and level of protective actions, regardless of exposure situation. ICRP considers that this may lead to an enhanced implementation of protection for interventions.

Radiation and tissue weighting factors and detriment figures are updated as indicated above, but ICRP emphasises that the changes to the risk factors are relatively small and the dose limits remain unchanged. Nevertheless, there is a warning that investigations concerning the lens of the eye are under way and may later warrant additional advice.

ICRP emphasises that the effective dose provides a measure of radiation detriment for protection purposes only. It does not provide an individual-specific dose and should not be used for epidemiological evaluations. Furthermore, the collective effective dose, the main use of which is in the optimisation of radiological protection, should not be used in epidemiological studies and in assessing the hypothetical number of cases of cancer or heritable disease in an exposed population.

Dose constraints and reference levels

Dose constraints restrict the range of options that should be considered in the optimisation process. In the case of public exposure, they provide for the possibility that a member of the public could be exposed to be a number of separate sources and still remain within the overall dose limit. A dose constraint might therefore be used by a regulatory body as the basis for establishing authorised limits for the discharge of radioactive material to the environment. In the case of occupational exposure, where workers are normally exposed to only one significant source, the dose constraint helps to focus the attention on good management of the exposure of workers, in the design of facilities, and in the planning of operations, and thus increases the equity. In emergency and existing exposure situations, when the prevailing conditions rather than any protection decision 'constrain' the possible individual doses, the corresponding planning level is called a 'reference level' rather than a 'constraint'.

Protection in medicine

'Diagnostic reference levels' are used in the context of the optimisation of the protection of patients undergoing medical exposure. These are intended to act as a benchmark figure against which doses from common diagnostic procedures can be compared.

For radiation therapy, optimisation of protection of the patient entails the reduction of doses outside the target volume for a given dose to the target, as well as reducing the probability of accidents.

Protection of the environment

The aims of protection are now extended to include the environment and in particular non-human species in their own right. The framework will be developed through the establishment of relevant data for a small set of reference animals and plants that are typical of 'the major environments'. At this stage, however, ICRP does not propose to set any form of 'dose limits' with respect to environmental protection.

Conclusion

A detailed description of the physical and biological information underpinning the Recommendations is provided in Annexes to the Recommendations. Additional advice directly pertaining to the Recommendations will be provided in reports on the scope of radiological protection and on radiological protection in medicine.

For radiological protection practice, the basic approach to normal planned work remains much the same as earlier, although the need to follow recommendations is expressed more stringently. However, in emergencies there are now more demands on the optimisation of protection.

Key note lecture. Science and Values in Radiological Protection – Impact on Radiological Protection Decision Making

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Abstract

This work summarises the main ideas and achievements of the Science and Values in Radiological Protection Workshop that was held on 15-17 January 2008 in Helsinki, Finland. In the view of developing of new radiological applications and emerging scientific phenomena it has been recognized a need to develop a shared understanding of emerging challenges for radiological protection among scientific and regulatory communities, public and other concerned stakeholders. In response to this the Committee of Radiation Protection and Public Health of the OECD Nuclear Energy Agency and Radiation and Nuclear Safety Authority of Finland tried to initiate a process of longer-term reflection on scientific and societal issues that might challenge radiological protection in the coming years.

Among general issues like radiological policy issues, improvement of understanding between research and policy communities, sharing views on emerging scientific issues, there were addressed several scientific issues, like non-targeted effects, individual sensitivity; and circulatory diseases. The main focus of these discussions was to elaborate potential "what if" scenarios and propose feasible solutions at various levels.

These discussions addressed effects that are not direct and evident consequence of the initial lesions produced at the cellular and DNA level like bystander responses, genomic instability, gene induction, adaptive responses and low dose hypersensitivity. Particular interest was paid to an extrapolation of risk estimates to low doses and role of Linear Non-Threshold theory in setting regulatory principles. Individual radio-sensitivity and identification of genes that are suspected of having an influence on it were also discussed in one of the Breakout Sessions. Another Breakout Session addressed circulatory diseases. There is emerging evidence in the A-bomb survivors and in other exposed groups that ionising radiation also causes other diseases than cancer, such as circulatory diseases. Elucidation of possible mechanisms is important, as there is also some evidence that chronic low-dose radiation may in fact be protective.

Each topic was addressed initially in plenary and subsequently in the parallel breakout sessions which were moderated by designated invited experts in the field. It is believed that the presented issues contributed to a more shared understanding between various stakeholders and integration of new scientific and technological developments and socio-political considerations into novel radiological protection.

KEYWORDS: radiological protection, science and values, non-targeted effects, circulatory diseases, individual radio-sensitivity

Introducdtion

During 2007, the CRPPH Expert Group on the Implications of Radiological Protection Science (EGIS) published a report on "Scientific Issues and Emerging Challenges for Radiation Protection" (1). This report represents a broad summary of key scientific challenges that could arise from ongoing research on radiological protection. Additionally, the CRPPH Expert Group on the Collective Opinion (EGCO) published its report on "Radiation Protection in Today's World - Towards Sustainability" (2). That report identifies key emerging challenges to the radiological protection system in order to assist decision makers at all levels to better address these within their relevant contexts.

Radiological protection is a combination of science and value judgments. There is a need for radiological protection policy makers, practitioners and other stakeholders to better understand the evolving interactions between science and values in the development of radiological protection policy and its practical application. At the same time, there is also a need for radiological protection scientists to better understand the broad processes of radiological protection decision making and to better interact with these processes in terms of furnishing input coming from their research.

Achieving mutual understanding among radiological protection policy makers, practitioners, researchers, industry, NGOs, etc. would facilitate prioritization of research and framing of decision making in the future and will also improve the quality and robustness of radiological protection.

Workshop objectives

The Committee for Radiation Protection and Public Health (CRPPH) has recognized a need to develop a shared understanding of emerging challenges for radiological protection among scientific and regulatory communities and other concerned stakeholders. In response to this the CRPPH has initiated a process of longer-term reflection on scientific and societal issues that might challenge radiological protection in the coming decade. The first step in this process was a workshop organized in collaboration with the Radiation and Nuclear Safety Authority of Finland (STUK) to address some of these issues. The workshop took place in January 2008 in Helsinki.

The objective of the workshop was to initiate a process of reflection and dialogue among the research community, policy makers and other stakeholders that would, in the longer term:

- improve understanding in both the research and policy communities on what is at stake in the system of radiological protection as scientific knowledge and social values evolve;
- contribute to the development of a more shared view of emerging scientific and societal challenges to radiological protection, taking into account existing differences;
- identify research that will better inform judgments on emerging issues;
- be the first step in the identification of elements of a framework that is better suited for the integration of new scientific and technological developments and socio-political considerations in radiological protection; and
- identify the most appropriate next steps in this process.

To achieve the above objectives, selected examples of emerging radiation protection issues were addressed in the workshop. Some of the key scientific issues identified in the EGIS report were used

as examples in the workshop, namely: non-targeted effects, individual sensitivity, and circulatory diseases.

Each topic was addressed initially in plenary and subsequently in the parallel breakout sessions which were moderated by designated experts. The moderated discussions followed a "what if" approach during which the nature and significance of the potential implications of the various emerging issues or challenges were addressed. Where appropriate, the need for further research and/or analysis was identified in order to better understand the challenge and how/if it may be accommodated. The outcomes of the breakout sessions were presented in plenary by the respective moderators followed by open discussion of each. The plenary lectures are available at NEA website (3)

Non-Targeted Effects: Issues and Discussion

Non-targeted effects refer to those effects that occur in cells not directly hit by ionizing radiation. In particular, what are called bystander effects are effects that occur in cells that were not traversed by radiation and are induced by signals from irradiated cells. Another significant not-targeted effect occurs in the genetic offspring of the irradiated cell, where an increased rate of genomic alterations is seen in the progeny of irradiated cells.

There is a substantial and growing body of knowledge in the area of non-targeted effects, and much research continues in this area. It is known that bystander effects can be induced in un-irradiated cells by ionizing radiation, and this has been seen in many experiments, including in-vivo experiments in a human skin cell model, in mouse experiments, and in experiments with blood samples from irradiated humans. Bystander effects are thought to be mediated by cell to cell gap junction communication or soluble factors, and they are seen at low doses (on the order of a few mGy) in situations where there are no direct irradiation effects from significantly higher doses (on the order of a few tens of mGy). Many types of effects can be generated in bystander cells, including gene expression, mutation, transformation, micronuclei, cell killing, DNA double strand breaks, γ-H2AX foci, chromosomal aberrations, micronuclei, or cell apoptosis. The link between these bystander effects and tumor genesis has not, however, been established.

Genomic instability, another type of non-targeted effect, is also known to occur as a result of irradiation. In this case, progeny cells which have not been irradiated manifest an instability in their genomic makeup, which can include chromosomal rearrangements, micronuclei, aneuploidy, delayed mutation (spectrum different), gene amplification or even cell killing. These effects can occur in cells that may be several generations beyond the originally irradiated cell. Here again, however, no link between genomic instability and radiation-induced carcinogenesis has been established.

Currently there is much that is unknown with regard to these two aspects of non-targeted effects. For example, in the area of bystander effects, it is not known what the chemical messengers are, that result in damage being manifest in non-irradiated cells. Nor do we know why damage occurs only in some cells in the vicinity of the irradiated cell. In the area of genomic instability, we do not know why this instability occurs irregularly in the family of progeny cells.

More generally, it is not known whether these effects are linked to the later appearance of diseases, such as cancer, leukemia, or circulatory diseases, and thus it is not known how these effects may affect the shape of the dose/response curve or the overall model of radiation-induced damage.

Why are non-targeted effects a relevant topic?

The precise nature of radiation-induced damage, and the mechanisms that lead to detrimental effects (i.e. cancer, leukemia, circulatory diseases, etc.) are not fully understood. However, we assume, for radiation protection purposes, that detriment is proportional to dose. If, however, effects in cells beyond those that are directly hit by ionizing radiation influence the genesis of radiation-induced diseases, this would suggest that some sort of amplification may be occurring and that, in fact, detriment is not as directly proportional to dose as we currently suspect. If this is, at some point in the future, scientifically demonstrated, while it is not clear that this would necessarily affect the practical implementation of radiological protection, it is clear that it would be necessary to reconsider the dosimetric criteria on which we currently base protection decisions and decide whether or not new or additional criteria or approaches would better guide protection decisions.

Although non-targeted effects are still couched in uncertainty, the above discussion suggests that it is valid to consider possible implications for radiological protection even at this early stage. There are many aspects of non-targeted effects that remain unknown or very uncertain, and knowing more about these aspects would facilitate taking better radiological protection decisions.

In addition to the scientific questions, the issue of non-targeted effects also raises a series of significant regulatory questions. Experiments with cell cultures would propose that the non-targeted effects may be relatively more important at low doses. However, we do not currently know if this would in fact change the health risk estimates. Broadly, at our current state of knowledge it is not clear whether these effects are biologically “good” or “bad”, making it difficult to understand how they might impact current radiological protection practices. For example, in medical treatments of patients, is the target area at risk really a greater volume than that actually irradiated? In general, do non-targeted effects amplify the detrimental effects of radiation, and if so how do we build non-targeted effects into radiation risk estimates? The fact that part of the radiation-induced effects may be induced only after the exposure might open new avenues for prevention of health detriment after exposure.

Overall, it was felt that better understanding of non-targeted effects would probably not affect the overall level of risk, but rather would better explain from where the risk originates. The non-targeted effects may also provide mechanistic explanations for development of other diseases than cancer. The non-targeted effects currently imply a change in the radiobiological paradigm but not a change in the radiation risk paradigm. Thus, it would not be necessary, based on our current, incomplete understanding, to change the approach that is currently taken to radiological protection to “better” protect people because of non-targeted effects.

Individual Sensitivity: Issues and Discussion

Individual sensitivity refers to the tendency of some individuals to be more or less sensitive than other individuals to radiation-induced damage. Such “hyper-sensitivity” or “hypo-sensitivity” can result

from genetic differences, but can also be affected by living conditions (i.e. environmental exposure to other toxic substances) or lifestyle choices (i.e. smoking).

Individual sensitivity is known to be expressed at "high doses", that is, levels experienced by patients undergoing radiation therapy, and may be expressed at "low doses", that is, exposures levels experienced by occupationally exposed workers and by the public in general. With respect to radiation therapy patients, as previously stated 5% are hyper-sensitive to radiation, and of these, 5% (or 0.25% of all therapy patients) are very hyper-sensitive. Importantly, it is also suspected that there some people who are hypo-sensitive to radiation, but the size of this group is not known.

Beyond these high-dose effects, much remains unknown about possible low-dose effects due to hyper-sensitivity. For example, what fraction of the population is hyper-sensitive to low-dose radiation, and how can their distribution(s) be characterized (i.e. in terms of gender, geographic location, age or habit data)? How much more radiosensitive is this group, and are there differences with type of radiation or with dose rate? Can the LNT model be reasonably used to characterize risks to these individuals, and if so, how would the risk factor(s) be determined? Does an individual's hyper-sensitivity to high doses imply that they are also hyper-sensitive to low doses, and if so, how can this be experimentally studied? Do lifestyle choices affect individual sensitivity? Can radiation hyper-sensitivity be linked to effects other than cancer and heritable effects, i.e. circulatory diseases?

Focusing on studies of effects at low-dose on humans, there is limited epidemiological evidence of stochastic effects below 100 mSv in adults, and 50 mSv in children. It is clear, however, that classical epidemiology has not and can not provide any evidence of individual sensitivity, mainly due to the need for extremely large samples to obtain statistical resolution of these effects.

Although this knowledge base is rather incomplete, it does suggest that further study is necessary to better understand the magnitude and nature of these possible effects.

Why is individual sensitivity a relevant topic?

The fact that some fraction of the population could be more or less sensitive to detrimental effects from ionizing radiation suggests that this is an issue that should be studied by governments and regulatory authorities. Individual sensitivity is plausible scientifically, and could result in significant risk differences. The significance of this to the management of radiological protection is that the current system of protection is based on a broad, averaged approach that applies equally to all exposed or potentially exposed individuals. As such, decisions regarding justification, optimization or limitation will not inherently account for variability in sensitivity, and thus may pose greater risks to some individuals than to others.

It has been known for some time that, on average, women are twice as sensitive to radiation-induced stochastic effects (mostly breast cancer) than men, and that, again on average, young children (that is those of about 5 years and under) are about five times as sensitive to radiation-induced stochastic effects as adults. Because of the relative lack of specific risk data on human populations, the UNSCEAR and the ICRP have judged that it is not appropriate to calculate age- and gender-dependent risk factors, but rather to use an androgynous, adult model as the basis for prospective exposure

management. By using dose as the primary quantity for exposure management, the ICRP in effect avoids the necessity of taking age and gender risk differences into account. In general, it is also true that risk differences of less than an order of magnitude are generally well within the calculational uncertainty of our current level of knowledge, so are often not judged to be significant by risk management experts. However, stakeholders may not feel that differences of a factor of two or five should be dismissed statistical noise.

High-dose considerations are particularly relevant because they are known to exist in radiation therapy patients. In particular, the link with specific genetic characteristics is being used to develop predictive tests that would indicate whether or not an individual would be likely to be hyper-sensitive to radiation. However, for such tests to be truly useful in helping to define an individual's treatment strategy it is important to better understand the mechanisms and consequences of effects caused by hypersensitivity, and their applicability, that is, at what range of exposures they might occur, what effect age at exposure may have, etc.

The issue of individual sensitivity raises a series of significant regulatory questions. For example, since hyper-sensitive individuals are included in the exposed populations that have been used as the basis for the estimation of radiological risk, in particular the populations of Hiroshima and Nagasaki, does this sufficiently take into account the risks hyper-sensitive individuals? In fact, is most of our current risk estimate actually due to risks in these individuals? If so, would it be appropriate to re-evaluate our current approach to radiological protection, either identifying a new dose limit to best protect hyper-sensitive individuals and another for "normal" individuals, or keeping a single dose limit but setting it as a function of risks to hyper-sensitive individuals.

An important challenge posed by our current level of knowledge is the need to assess what changes would need to be made in our current radiological protection approach when knowledge evolves. Adopting a "what if" approach, several changes can be foreseen once:

- A tool exists to prospectively identify or predict individual sensitivity;
- An understanding of the fraction of the population that is more sensitive is developed, and of their relevant distribution(s);
- An understanding has been developed of how much more sensitive the population is;
- Knowledge of the relationship between sensitivity to acute effects and to stochastic effects has been sufficiently developed;
- Knowledge of low-dose and dose-rate effects, whether negative, positive or neutral has been sufficiently developed

Based on this level of understanding it is likely that radiological protection changes would be considered for both high- and low-dose situations (therapy, triage, emergency workers, and specific groups). These approaches would necessitate serious consideration of various ethical and equity issues, insurance coverage etc. In addition, there would be a need to provide education and information on these issues.

Based on our current level of knowledge, in particular on our understanding of the probable levels of increased risk should large populations of hyper-sensitive individuals exist, there seem to be no need to radically modify the current approach to radiological protection. In three exposure situations, however, it was suggested that some consideration should be given at this point to refocusing protective actions taking individual sensitivity into account. These are emergency exposure situations

(children and pregnant women), medical diagnosis situations (optimization and awareness of patient doses, especially for CT), and medical therapy situations (awareness of individual sensitivity and of secondary cancers).

Circulatory Diseases: Issues and Discussion

It has been generally accepted that high dose (several Gy) radiation exposure to the heart or other parts of the circulatory system result in long-term increases in circulatory disease risks. Over the past 10-15 years evidence has been emerging from the long term follow-up of atomic bomb survivors and other populations that relatively low dose acute exposures (< 2 Gy) are also associated with increased circulatory disease risks. Although the estimated relative risks are smaller than for cancer, it is clear that radiation-associated circulatory disease deaths will account for a substantial fraction of the total radiation impact on mortality in the atomic bomb survivors. However, those epidemiological data do not, and probably cannot, provide definitive evidence of increased circulatory disease risks following low dose (say 0.005 to 0.5 Gy) exposures. Despite this uncertainty, these findings have increased interest in efforts to identify mechanisms for long-term radiation effects on the circulatory system and prompted the re-examination of circulatory disease risks in other populations.

As far as the potential mechanisms of the radiation induced cardiovascular diseases are concerned, there are several hypotheses (inflammatory, micro vascular, mutation induced, and others). An inflammatory mechanism that is more consistent with deterministic effects is currently more plausible. Several questions that need to be addressed in this respect are:

- Are there different mechanisms at high and low doses?
- Are these mechanisms consistent with stochastic or deterministic dose response?
- What is the threshold? If the threshold is low, there may be a need for change in radiation protection.
- Does the relative risk depend on type of cardiovascular disease?
- How does the spectrum of radiation induced cardiovascular disease depend on dose?
- What is the link between cardiovascular disease and dose, dose-rate effect, and radiation quality?
- What are the age, gender, population and temporal effects?
- What is the importance of synergistic effects, i.e. interactive effects with other agents?
- What is the target tissue?

Why are circulatory diseases a relevant topic?

Circulatory diseases are currently not specifically addressed by the radiation protection system. The ICRP recognizes the existence of this problem, however it notes that experimentally observable dose associated effect is at high doses, around 1 Gy. There are still uncertainties on the shape of the dose-response at low doses and whether these effects have a threshold at around 0.5 Gy or whether there is no threshold. In general, it accepts that available data do not allow for their inclusion in the estimation of detriment following low radiation doses less than 100 mSv. This also agrees with the conclusion of UNSCEAR 2008 which found little evidence of any excess of risk below 1 Gy.

The new UNSCEAR report includes an annex on this topic and it seems inevitable that the ICRP and other groups involved in the formulation of regulatory guidelines will have to deal with the question of how to incorporate potential circulatory disease risks into the evolving system of radiation protection.

The Breakout session concluded that if potential changes in radiation protection principles are made based on available Japanese risk estimates and LNT, there will be a significant need for revision of the radiation protection criteria. These would need to lower current dose limits by 30-50% with strong emphasis on optimization. The application of precautionary principle should include not only the change in detriment but also the cost and other consequences associated with this change. If this is the case, the current radiological system will be significantly challenged. However, the Session also recognizes that any potential change shall be made in the light of evolving science and serious value judgments.

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2 Medical session

Current activities of the Nordic Working Group on X-ray diagnostics

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Abstract

Introduction: A formal and practical co-operation between the Nordic radiation protection authorities has existed for decades. The co-operation has been mainly implemented through Working Groups (WG) established for several fields of interest. The WGs have sometimes been established for specific tasks, but usually they have mandates for more permanent working on a specific topic area. At present there are altogether eight working groups in operation. The main tasks of the Nordic WG on X-ray diagnostics are to exchange information on the national activities and achievements, to discuss problems and development for regulatory issues and to coordinate or undertake joint efforts of research. The joint research projects aim at supporting the regulatory activities and promoting the development of quality assurance and radiation safety in diagnostic radiology. The results of the work are published in the series of the Report on Nordic radiation protection co-operation (seven reports by 1999) and also as scientific papers in international journals.

Material and method: The current work of the Nordic WG on X-ray diagnostics is summarized.

Results and discussion: The current members of the WG on X-ray diagnostics are the authors of this presentation. The WG meets at least once a year in different Nordic countries in turn. The chairman and secretary serve in three year turns and the turn is circulated amongst the Nordic countries. The WG submits annually a report on its work to the meeting of the Chiefs of the Nordic radiation protection authorities.

In the WG meetings, the recent developments in each Nordic country, for a large number of topics, as well as the joint research projects and other joint activities are reviewed and discussed. Besides administrative and statistical information, the topics recently discussed include diagnostic reference levels, issues on patient dosimetry, quality assurance of x-ray equipment, special applications (e.g. computed tomography and interventional radiology), diagnostic dosimetry and clinical audit. A recent joint project, Nordic survey on paediatric CT examinations, is presented in another paper of this NSFS meeting. Plans for new projects include studies on justification of CT

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examinations and the implementation of the optimization principle in x-ray diagnostics. Other joint activities include workshops on inspections, now organized two times by the Swedish counterpart (SSI), plans for IRPA 2010 meting in Helsinki and cooperation with the Baltic countries in the field of x-ray diagnostics.

The annual meetings, joint projects and other joint activities of the WG have ensured effective change of information between the Nordic authorities in the field of diagnostic radiology. This has been of high importance for developing the national regulatory activities, for promoting consistent methods of quality assurance and dosimetry and, in general, for improving the safety and efficacy of diagnostic radiology.

KEYWORDS: X-ray diagnostics, Nordic co-operation

Comparison of technical performance between CBCT and low-dose MDCT for oral and maxillofacial radiology

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Abstract

Aim: The Institute of Clinical Denstistry, University in Oslo, has had a cone beam CT (CBCT: Promax 3D, Planmeca OY, Helsinki) for half a year to test its clinical use for maxillofacial radiology. Department of Maxillofacial Radiology has an eight slice multi detector CT (Lightspeed Ultra, GE, MI), and wanted, as a collaboration project with the Norwegian radiation protection authority, to evaluate and compare the technical performance of the two modalities in a clinical setting with respect to image quality and the radiation dose to patients examined.

Material and method**:** The patient dose in CBCT was measured with a calibrated DAP meter (PTW, Freiburg), while the doses in MDCT was calculated by means of the CT dosimetry spreadsheet (Impact, London). The image material was gathered from one single patient who voluntarily went through examinations with both CBCT and MDCT.

Results: Good agreement between the measured DAP values and the values indicated on the CBCT console after the exposure was found. The effective doses from CBCT are in the range 0,04 – 0,27 mSv for various patient sizes and field of view (FOV). MDCT protocols particularly optimised for low dose oral and maxillofacial radiology gave doses at the same range. The last version of the CBCT software gave more artefacts compared to the previous version, and did not manage to reconstruct the "lower" FOV properly.

Discussion/conclusion: There seems to be some initial problems with the cone beam reconstruction algorithm with this particular machine, possibly combined with the equipment movement/raw data registration. The images based on the previous software were, however, quite good and rather similar to the CT images of the same anatomical region of jaw bone. The patient doses were also similarly low with both modalities. The measurements and quantification of noise and artefacts in CBCT will have to be evaluated further, to be able to draw fair comparisons.

Introduction

Computed tomography (CT) was introduced during the 1970's, and has developed from axial single slice scanners, into helical technology and the currently used multi detector scanners (MDCT). The recent development of a cone beam reconstruction algorithm has been utilized in the so called cone beam CT technology (CBCT), that involves applications both in radiation therapy (units installed as a part of the linear accelerator to image the target volume) and in oral and maxillofacial radiology.

The Department of Maxillofacial Radiology, University in Oslo, has had a CBCT (Promax 3D, Planmeca OY, Helsinki) for half a year to test its clinical use for maxillofacial radiology. The department also has an eight slice multi detector CT (Lightspeed Ultra, GE, MI), and wanted, as a collaboration project with the Norwegian radiation protection authority (NRPA), to evaluate and compare the technical performance of the two modalities in a clinical setting with respect to the radiation dose to patients examined and also the image quality.

The Promax 3D unit is based on a traditional Orthopantomogram (OPG) unit, but the film cassette is exchanged with a digital detector plate. The patients head is positioned between the X-ray source and the image receptor. The attenuated radiation is registered during a 180º tube rotation, and cross sectional slices of the volume of interest are reconstructed by a so called"proprietary Feldkamp based back projection". Thus, even though the name CBCT associates to a CT scanner, it is the reconstruction algorithm that is the common feature, not the apparatus as such (Figure 1).

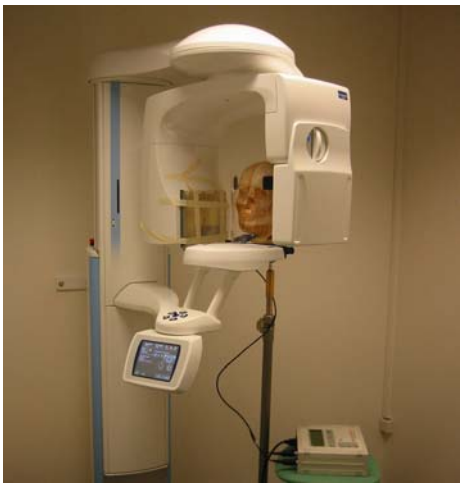


Figure 1*:*

The Cone beam CT (Promax 3D, Planmeca OY, Helsinki) with a phantom positioned to illustrate the patient. A DAP meter (PTW, Freiburg) is mounted on the tube side to measure the patient dose.

Effective dose is a calculated quantity used in radiation protection to say something about the probability of late effects, and involve both the knowledge about organ doses and the associated organ specific radiation risk. The International Commission on Radiological Protection (ICRP) defines the effective dose according to latest presumptions about radiation risk. The dose calculation programmes used currently are based on the 1990 ICRP recommendations [1]. However, the effective dose has been redefined in the recently published 2007 ICRP recommendation [2], which will give some different figures. Anyhow, the effective dose is the only applicable dose quantity when the aim is to compare different modalities such as MDCT, CBCT or conventional radiographic examinations with respect to patient doses.

Ngan et al (2003) provide some figures for the effective dose (ICRP 1990 [1]) resulting from oral and maxillofacial radiology [3]; dental radiographs gives typically 0.005 mSv (intra-oral periapical radiographs) or 0.007 mSv (maxillary occlusal), cephalography (lateral cephalometric radiographs) gives about 0.005 mSv, while an OPG examination results in 0.010 mSv. Computed tomography examinations (CT) give normally higher doses in comparison; a maxillo-mandibular CT scan gives 2.1 mSv, a maxillary CT scan gives 1.40 mSv, while a mandibular CT scan gives 1.32 mSv. These reported dose values from CT are in agreement with published dose data from Norway [4].

Ludlow et al (2006) provide dose data for three different CBCT manufacturers, and have calculated the effective dose both according to former (still used) and recent definition [1, 2]. The figures in units

of mSv are (E_{1990}, E_{2007}): NewTom3G (QR, Verona, Italy) (0.045, 0.059), i-CAT (Imaging Sciences International, Hatfield, PA) (0.135, 0.193), CB Mercury (Hitachi Medical Systems America, Twinsburg, OH) (0,477, 0,588), and as comparison for an OPG (0.0063, 0.0133) [5].

From the above results the following aspects are worth noticing:

- The CBCT technology applied in oral/maxillofacial radiology seems to give doses in the range 4 – 50 times higher than a traditional OPG examination, but still significantly lower compared with a MDCT scan
- Different CBCT's reveals a manufacturer dependent variation in doses of almost a factor of ten, probably due to underlying differences in technology, detector sensitivity, FOV, etc
- Regarding the maxillofacial region of the body, the effective dose values calculated according to the 2007 ICRP recommendations [2] are generally higher compared to what would be the case using the 1990 recommendations [1]. This is probably due to upward revision of the weighting factors for brain and salivary gland, and also the calculation of "remainder" organs

The aim of this pilot study was to do some tests of the Promax 3D unit; to compare the measured DAP values with the figures provided on the operators consol after an examination, and to compare the effective dose associated to the use of CBCT with other published results and two suggested low dose MDCT protocols at the Department of Maxillofacial Radiology in Oslo. Another future aim is to identify or develop some quantities or criteria that may be used for evaluation of image quality both in CBCT and MDCT for comparison of the two modalities. For the latter, only the interpretation of image noise is discussed briefly in this paper.

Material and method

Plane parallel ionisation chambers mounted on the tube diaphragm housing, so called DAP meters, may be used for both dental radiographs (bite wing), OPG and CBCT [6]. A DAP meter (PTW, Freiburg) calibrated for the relevant X-ray energies at NRPA's secondary standard laboratory was used for the measurements. Conversion coefficients between dose area product and effective dose were provided from the research community in Linköping, Sweden [7]. The CBCT doses were measured for some optional fields of views (FOV 80mm, 50mm "upper" and 50mm "lower") in October 2007. The doses in MDCT was calculated for the actual scan parameters used in the clinic by means of the Excel based CT dosimetry spreadsheet developed by the Impact CT evaluation centre at St. George's Hospital in London [8]. This tool is based on conversion coefficients provided by Health Protection Agency (HPA) in UK (former NRPB), resulted from Monte Carlo simulations [9]. None of these tools currently provide options for the calculation of effective dose according to the new ICRP2007 recommendations [2].

According to the technical specifications for the CBCT (Promax 3D, Planmeca OY, Helsinki), the detector resolution is 624x624 pixels, the pixel size is 200μm, the number of voxels are 501x501x501=125M, and the voxel size is isotropic; 160x160x160 μm. The object for reconstruction in MDCT is the value of the linear attenuation coefficient in each single voxel. It is assumed that this is the case also for the CBCT. In conventional CT the voxel values are normalized to water in terms of CT-numbers (Hounsfield units, HU). It is not clear to the authors exactly how this is applies to CBCT, and furthermore, how the calibration curves would look like. Anyhow it is expected that the voxel values or "grey values" for the same kind of material may differ between CBCT and MDCT since the tube voltage during acquisition is different.

The image material was gathered from one single patient who voluntarily went through examinations both with CBCT and MDCT in May 2007. Apparently the patient was oriented somewhat differently with respect to the axis of rotation, but we tried to select about the same cross section for comparison of the MDCT and CBCT images. The Feldkamp back projection in CBCT is based on 0.16mm isotropic voxels. The reconstruction software (Romexif version 1.5.1.0.R, Planmeca OY, Helsinki) provides only isotropic reformatting of the images. In order to compare the CBCT images with the 2x0.625mm axial MDCT reconstructed images we therefore used the spatial slice averaging filter of the free software package MIPAV (Version 4.0.3, Medical Image Processing, Analysis and Visualization, National Institute of Health, US http://mipav.cit.nih.gov/). The images were viewed using ImageJ on a PC (freeware http://rsb.info.nih.gov/ij/). A region of interest (ROI) was positioned in the region of *musculus genioglossus* both in the CBCT and the MDCT images (figure 2 and 3 behind), and the ROI statistics were presented in histograms. The standard deviation of the pixel values in the ROI was used as a measure of image noise. The patient images were also analysed subjectively with respect to artefacts.

Results

There was between 2 – 4% deviations between the measured DAP values and the figures indicated on the CBCT console after the exposure for the 80mm FOV (good agreement). The effective doses from CBCT were in the range 0,06 – 0,43 mSv for the patient sizes and fields of views (FOV) that were measured. The dose values for the single patient examined both with CBCT and MDCT are presented in Table I. Cross sectional images of this patient are shown in Figure 2 and 3, also showing the statistics for the ROI values. The CBCT used gave doses equal to the 90mA MDCT protocol for this particular patient. The CBCT images and the images provided from the 90mA MDCT scan also presented very similar values of image noise (sd = 67 and 61HU, respectively), while the absolute voxel values were different. The corresponding result of image noise in the CBCT image based on the averaging of four images resulting in 0.62mm slice thickness had sd=44.

Table I: *Comparison of scan parameters, patient dose and image noise for a CBCT (Promax 3D, Planmeca OY, Helsinki) protocol and two MDCT (Lightspeed Ultra, GE, MI) protocols used on the same single patient*

	CBCT	MDCT low dose I	MDCT low dose II
Scan parameters	84kV, 12mA (6s), 80mm FOV isotropic voxel size; 160x160x160 µm beam height 50mm ”lower”	120 kV **90 mA** 0.8sec 2x0.625mm slice thickness, Axial 2i, speed 1.26mm/rot, 1:1 6 cm scan length	120kV **20mA** 0.8sec 2x0.625mm slice thickness, Axial 2i, speed 1.26mm/rot, 1:1 6cm scan length
Dose figures	DAP=1.536 Gycm^2	$CTDI_{vol}$= 20,5 mGy DLP= 123 mGycm	$CTDI_{vol}$= 4,6 mGy DLP= 27 mGycm
Effective dose [i)]	0.20mSv	0.20mSv	0.045mSv
Mean (voxel values) [ii)]	– 198	65 HU	62 HU
Sd (voxel values) [ii)]	67	61 HU	126 HU

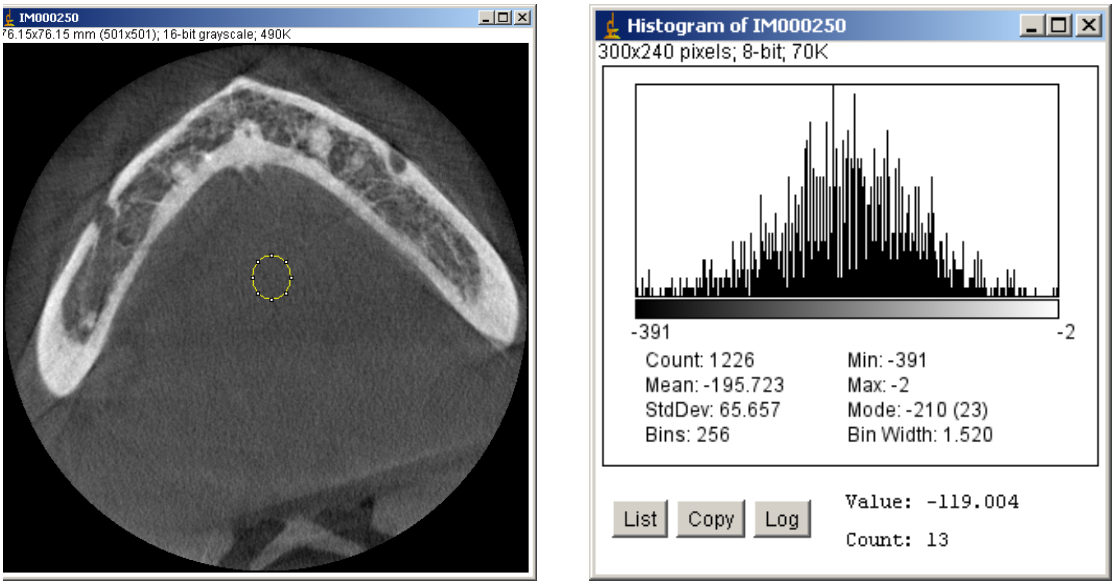
i) For the effective dose calculation in CBCT the conversion factor 0.13mSv/Gycm^2 was used [7], while for the effective dose calculation in MDCT it is presumed the tyroidea is not included in the scan volume

ii) The ROI is positioned in the region of *musculus genioglossus* (see also Figure 2 and 3)

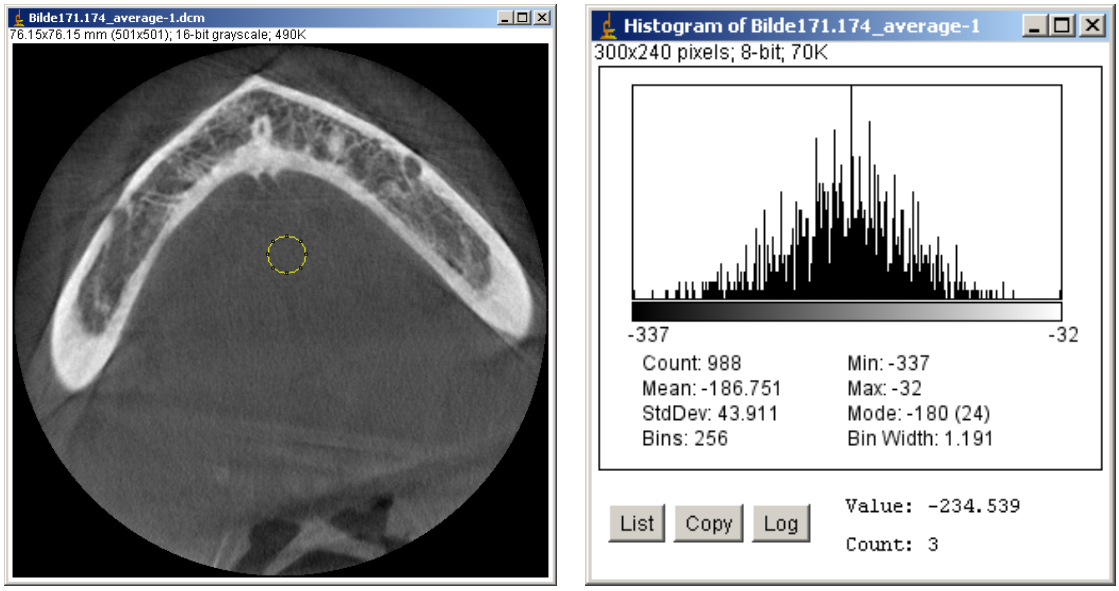
Discussion / conclusion

According to the conventional CT theory, the image noise (standard deviation of the voxel values in a ROI) should be inverse proportional to both the mAs product and the reconstructed slice thickness. From the ROI value in the 90 mA MDCT image we may calculate the expected noise in the 20mA image as: sd= 61/sqrt(20/90)=128HU. The measured sd of 126HU is therefore very well in agreement with the theory (Figure 3). If this theory is applicable for CBCT, we would expect that the noise level would halve when four CBCT images are averaged (Figur 2 a and b), i.e. sd=67/2=34 in the 0.61mm slice thickness image. The measured value of 44 is some higher, but within the uncertainty introduced by the position of the ROI's in the region of *musculus genioglossus* (inhomogeneous mixtures of material). Anyhow, between CBCT and MDCT, the reconstruction algorithms and the X-ray radiation qualities are different; the interpretation of the slice thickness, the CT number and the quantification of image noise may also be different. The interpretation of the results has to be investigated further in a larger material before certain conclusion can be drawn.

In oral and maxillofacial radiology many anatomical features have high subject contrast; that means the image noise is not a big issue whilst high spatial resolution is desirable. In such cases MDCT protocols may be optimized to give patient doses in the same order of magnitude as the CBCT technology. On the other hand the initial costs for a MDCT are higher. Even though the voxel size in CBCT is very small, it seems like the image noise level is about the same as in the low dose cross sectional MDCT images and the spatial resolution is certainly promising. The CBCT images based on the May 2007 software seemed rather similar to the MDCT images of the same anatomical region of jaw bone. There seems to be some initial problems with the cone beam reconstruction algorithm with the particular machine used for this comparison, possibly combined with an inaccurate equipment movement during raw data acquisition. In the further work on this topic we will develop image quality criteria applicable for oral and maxillofacial radiology with the EU CT quality criteria as the starting point (http://www.msct.eu/CT_Quality_Criteria.htm), which may be used both for MDCT and CBCT. Basically the MDCT and CBCT images look different, but if only segments of the images are presented for the readers, it may be possible to design blinded studies for evaluation of the two modalities.



Figur 2 a: *Pasient scanned with CBCT at 84kV, 12mA, 80 FOV, beam height 50 "lower", isotropic voxel size 160 μm*



Figur 2 b: *Pasient scanned with CBCT at 84kV, 12mA, 80 FOV, beam height 50 "lower", using the spatial slice averaging to give a slice thickness of 0.61mm (pixel size 160μm)*

Figure 3: *The MDCT images based on 0.625mm slice thickness (90mA upper versus 20mA lower)*

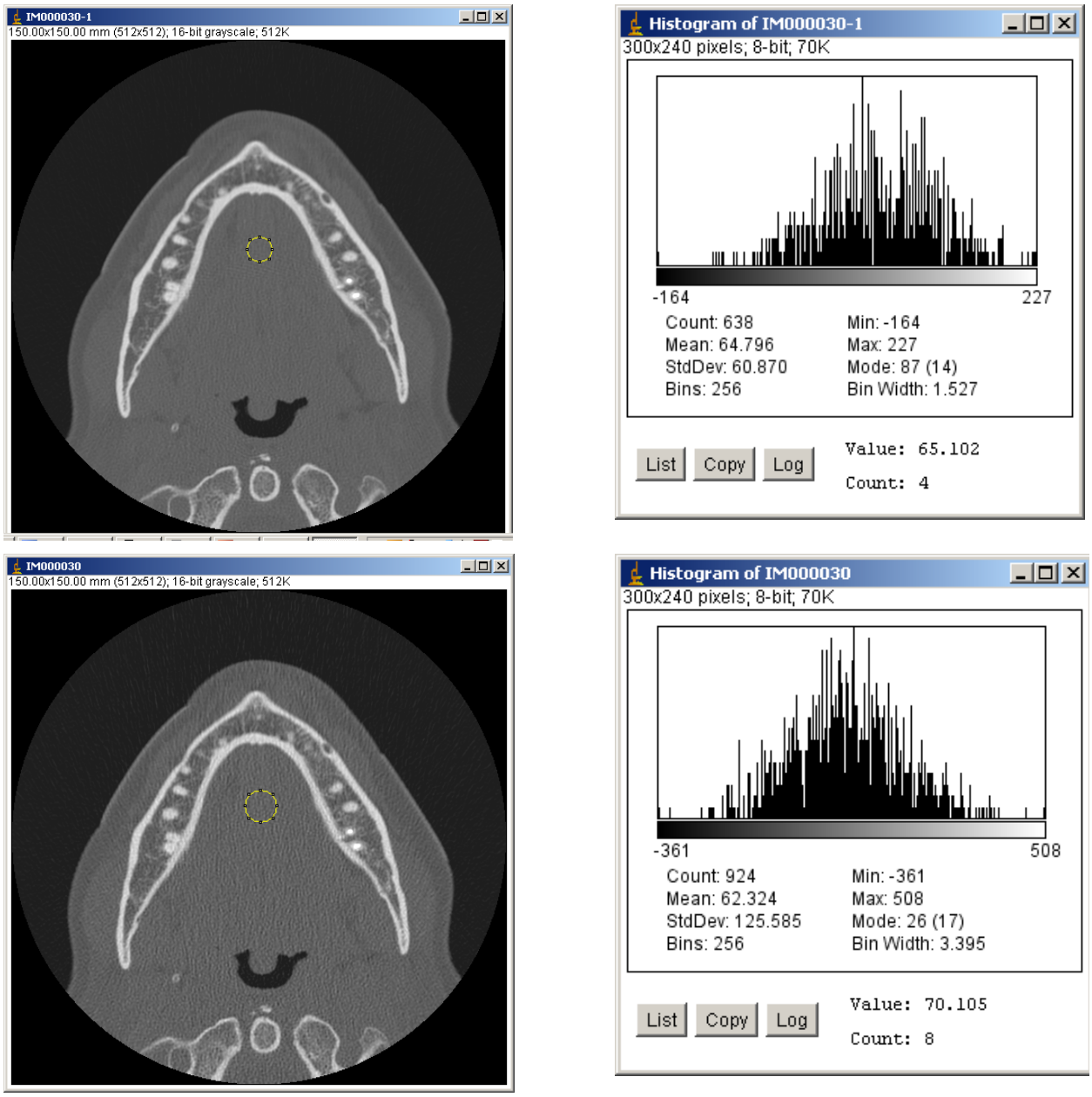


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Table 2: *Data from the QA measuring protocol of the CBCT (Promax 3D, Planmeca OY, Helsinki) (October 2007)*

	Tube voltage	Tube current	FOV (width)	Beam height preset	Beam height measured *)	DAP monitor	DAP measured	Calibration factor N_{KA}	DAP corrected	Conv factor [7]	Effective dose [1]
	kV	mA	mm	mm	mm	$mGycm^2$	$Gycm^2$		$Gycm^2$	$mSv/Gycm^2$	mSv
Adult female	84	12	80	80	70.8	2402	2.096	1.1682	2.449	0.13	0.32
Adult female	84	12	80	50(upper)	47.4	1461	1.273	1.1682	1.609	0.13	0.19
Adult female	84	12	80	50(lower)	18.3	1461	1.377	1.1682	1.461	0.13	0.21
Adult female	84	12	40	50(upper)	47.4	718	0.676	1.1682	0.790	0.13	0.10
Adult female	84	12	40	50(lower)	18.1	718	0.722	1.1682	0.843	0.13	0.11
Large Adult	84	16	80	80	76.6	3203	2.855	1.1682	3.335	0.13	0.43
Large Adult	84	16	80	50(upper)	47.8	1948	1.666	1.1682	1.946	0.13	0.25
Large Adult	84	16	80	50(lower)	18.4	1948	1.837	1.1682	2.146	0.13	0.28
Large Adult	84	16	40	50(upper)	47.5	958	0.894	1.1682	1.044	0.13	0.14
Large Adult	84	16	40	50(lower)	18.4	958	0.966	1.1682	1.128	0.13	0.15
Child <6 years	80	8	80	80	76.6	1478	1.289	1.1682	1.505	0.13	0.20
Child <6 years	80	8	80	50(upper)	47.8	899	0.757	1.1682	0.884	0.13	0.11
Child <6 years	80	8	80	50(lower)	18.4	899	0.831	1.1682	0.970	0.13	0.13
Child <6 years	80	8	40	50(upper)	47.7	442	0.406	1.1682	0.474	0.13	0.06
Child <6 years	80	8	40	50(lower)	18.3	442	0.438	1.1682	0.511	0.13	0.07

*) The actual machine had difficulties with reproducing the 50mm "lower" FOV in October 2007, while this option was well functioning in May 2007 when the patient images were gathered. The software had been upgraded in the meantime; these two circumstances lead to more artefacts in the images, particularly those using the "lower" FOV.

Table 3: *The Impact CT dose calculator spreadsheet out print for the 16 mAs protocol. The 72 mAs protocol will have correspondingly higher dose values*

ImPACT CT Patient Dosimetry Calculator

Version 0.99x 20/01/06

Scanner Model:			
Manufacturer:	GE		
Scanner:	GE LightSpeed Ultra		
kV:	120		
Scan Region:	Head		
Data Set	MCSET20	Update Data Set	
Current Data	MCSET20		
Scan range			
Start Position	76	cm	Get From Phantom Diagram
End Position	82	cm	
Patient Sex:	f		

Acquisition Parameters:			
Tube current	20	mA	
Rotation time	0.8	s	
mAs / Rotation	16	mAs	
Collimation	2x.63	mm	
Slice Width	0.63	mm	
Pitch	1		
Rel. CTDI	Look up	1.27	at selected collimatio
CTDI (air)	Look up	44.5	mGy/100mAs
CTDI (soft tissue)		47.6	mGy/100mAs
$_nCTDI_w$	Look up	28.5	mGy/100mAs

Organ	w_T	H_T	$w_T.H_T$
Gonads	0.2	0	0
Bone Marrow (red)	0.12	0.11	0.014
Colon	0.12	6.1E-06	7.4E-07
Lung	0.12	0.013	0.0015
Stomach	0.12	0.00036	0.000044
Bladder	0.05	0	0
Breast	0.05	0.0028	0.00014
Liver	0.05	0.00083	0.000041
Oesophagus (Thymus)	0.05	0.011	0.00055
Thyroid	0.05	0.36	0.018
Skin	0.01	0.12	0.0012
Bone Surface	0.01	0.48	0.0048
Remainder 1	0.025	0.1	0.0026
Remainder 2	0.025	0.1	0.0026
Total Effective Dose (mSv)			**0.045**

Remainder Organs	H_T
Adrenals	0.001
Brain	0.3
Upper Large Intestine	0.000038
Small Intestine	0.000026
Kidney	0.00018
Pancreas	0.00071
Spleen	0.00068
Thymus	0.011
Uterus	0
Muscle	0.1

$CTDI_w$ (mGy)	4.6
$CDTI_{vol}$ (mGy)	4.6
DLP (mGy.cm)	27

Scan Description / Comments	Pasient A undersøkt 10.05.2007 på UiO, Odentologisk fakultet

Doses from pediatric CT examinations and level of optimization of the scan protocols in the Nordic countries

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Abstract

Introduction: Doses to pediatric patients from CT examinations are known to be unnecessarily high if scan protocols developed for adult patients are adopted. Use of optimized size-specific pediatric scan protocols is essential to keep the doses at an appropriate level. The aim of this study was to estimate doses to pediatric patients from CT examinations and to evaluate the level of optimization of the scan protocols used in the Nordic countries.

Material and method: Applied scan parameters, clinical indication and patient data were collected for four examination areas (brain, chest, abdomen and whole body) from totally 20 hospitals within the Nordic countries. The available dose data on the scanner console, CT Dose Index corrected for pitch ($CTDI_{vol}$) and Dose Length Product (DLP) was also collected. Effective doses were estimated by multiplying DLP with conversion factors (published by Chapple et al, 2001) available for four separate scan areas (head/neck, chest, abdomen and pelvis). To best represent a pediatric body, the DLP for a standard head phantom (diameter 16 cm) was assessed and used for the effective dose estimates.

Preliminary results and discussion: Totally 359, 272, 192 and 73 scan protocols were received for pediatric CT examinations of respectively the brain, chest, abdomen and whole body. Preliminary calculations of the range (min-max) of the reported consol $CTDI_{vol}$ were 7.4-140.0, 1.6-15.1, 1.7-25.5 and 1.7-55.0 mGy for examinations of respectively the brain, chest, abdomen and whole body, while the corresponding range in consol DLP were 7-1783, 6-801, 29-1054 and 54-2144 mGy·cm. The age of the patients examined varied from 0 to 16 years and the most common clinical indications were trauma and tumor or controls of these indications. The mean effective dose for examinations of the brain, chest, abdomen and whole body was 2.9, 2.1, 4.4 and 5.5 mSv, respectively. Finalizing the dose calculations from all countries, the results will give valuable information of the dose levels to pediatric patients in the Nordic countries, especially when they are analyzed with respect to age, sex and clinical indication. When combined with the underlying scan parameters, the degree of optimization of the scan protocols applied to pediatric patients will be evaluated. Establishment of some common Nordic diagnostic reference levels for pediatric CT examinations will be considered when the dose results are finalized.

KEYWORDS: Pediatric, CT examination, optimization, scan protocol, effective dose, dose length product (DLP).

Introduction

Doses to pediatric patients from CT examinations are known to be up to three times higher if scan protocols developed for adult patients are adopted without adjusting the scan parameters according to the patient's size [1-3]. Development and use of optimized size-specific pediatric scan protocols is therefore essential to reduce the dose to pediatric patients. The aim of this study was to estimate doses to pediatric patients from CT examinations of four different body regions and to evaluate the level of optimization of the scan protocols among the Nordic countries.

Material and method

The dosimetric quantities volume weighted CT dose index ($CTDI_{vol}$), dose length product (DLP) and effective dose (E) were used as dose indicators in this study. Key data on local CT practice were collected by means of a questionnaire for examinations of the head, chest, abdomen and whole body. Information about the patient (sex, age, weight and height), clinical indication, applied scan parameters together with the available dose data displayed on the scanner console, $CTDI_{vol}$ and DLP, were among the collected data.

Effective doses to pediatric patients were estimated from the total DLP related to the 16 cm diameter phantom (DLP_{16cm}) by using region-specific conversion coefficients (E_{DLP}) published by Chapple et al [4]. These conversion coefficients were expressed as a function of patient size, making them applicable to all patient sizes. If the patient's weight and height were missing in the questionnaire, average weight and height for the given age and gender, were used in the calculation of E_{DLP}. DLP_{16cm} was obtained by different methods within the Nordic countries. Iceland calculated DLP_{16cm} from the collected scan parameters and own weighted CTDI measurements ($CTDI_w$) in the 16 cm diameter dosimetry phantom, while Finland obtained the DLP_{16cm} by direct measurements of DLP_{16cm} in the same phantom. Norway and Sweden used the reported DLP values displayed on the scanner consol. For examinations of the pediatric trunk these values were converted to DLP_{16cm} by multiplying with the ratio between normalized $CTDI_w$ measured in the 16 cm and 32 cm diameter dosimetric phantoms. The ratio between normalized CTDI measurements free in air ($CTDI_{air}$) for body and head scan field of view (SFOV) was also applied to account for potential differences in filtration between these two SFOVs. Scanner-specific normalized $CTDI_w$ and $CTDI_{air}$ measurements provided by ImPACT were used in this conversion [5]. Denmark did not report any effective doses.

Results

Totally 421, 222, 198 and 77 questionnaires were received for pediatric CT examinations of the head, chest (not including HRCT), abdomen and whole body, respectively, from 19 hospitals within the Nordic countries. Seventy of the questionnaires were filled out incorrectly or incompletely and were excluded from the survey. All countries received most questionnaires for head examinations, while questionnaires for examinations of the trunk were more modest. Iceland and Denmark received too few questionnaires for examinations of the pediatric trunk (n=19) and were also excluded due to poor statistics. The total number of questionnaires included for the different scan areas from each country and the number of participating hospitals are given in Table 1. All major CT vendors were represented in this study, covering totally 18 different CT scanner models (all helical). All the scanners, except one, were multi slice scanners giving 2 (n=1), 4 (n=5), 8 (n=4), 10 (n=1), 16 (n=12) and 64 (n=2) simultaneous slices per rotation.

Mean effective dose to pediatric patients (newborn-16 years old) in the Nordic countries were 3.1, 1.9, 5.3 and 5.4 mSv for examinations of the head, chest, abdomen and whole body, respectively. Analysis of the dose parameters $CTDI_{vol}$, DLP and effective dose with respect to specific age groups are presented in Table 2 and 3. Table 2 show the mean $CTDI_{vol}$, DLP and effective dose from all examinations from all the Nordic countries, while Table 3 shows the mean $CTDI_{vol}$, DLP and effective dose reported from each Nordic country.

Table 1: *Number of questionnaires for examinations of the different scan areas included from the Nordic countries. Total number of participating hospitals from each country is given in brackets.*

Country	**Norway**	**Sweden**	**Finland**	**Iceland**[1]	**Denmark**[1]	**Total**
Scan area	**(n=5)**	**(n=5)**	**(n=5)**	**(n=1)**	**(n=3)**	**(n=19)**
Head	134	121	55	28	56	394
Chest	84	70	42	-	-	196
Abdomen	80	79	16	-	-	175
Whole body	15	29	20	-	-	64

[1] Iceland and Denmark received too few questionnaires for examinations of the pediatric trunk to be included in the survey.

Table 2: *Mean values of $CTDI_{vol}$ [mGy] and DLP [mGycm] displayed on the CT scanner consol together with the mean effective dose (E) [mSv] for examinations of the head, chest, abdomen and whole body from all Nordic countries. In case of multiple scan sequences, the maximum $CTDI_{vol}$ and the total DLP for the examination is given. Mean age of the patients within each age group are given in brackets. $CTDI_{vol}$ and DLP are related to 16 cm diameter phantom for head examinations and 32 cm diameter phantom for examinations of the trunk. Effective dose are calculated from DLP_{16cm}.*

Scan area		**Head**			**Chest**			**Abdomen**			**Whole body**		
Age		**$CTDI_{vol}$**	**DLP**[1]	**E**	**$CTDI_{vol}$**	**DLP**[1]	**E**	**$CTDI_{vol}$**	**DLP**[1]	**E**	**$CTDI_{vol}$**	**DLP**[1]	**E**
0-1	(0.1)	27.0	394	5.5	2.7	36	1.4	7.7	284	7.2	-	-	-
1-2	(1.1)	29.9	472	3.3	4.8	66	1.9	4.2	131	5.4	-	-	-
2-5	(3.0)	37.0	558	2.3	2.9	50	1.4	3.9	137	4.3	8.7	327	4.1
5-10	(7.1)	45.5	614	2.3	4.0	80	1.8	5.8	246	4.7	10.6	538	5.9
10-16	(12.5)	55.5	580	2.3	5.6	158	2.9	7.4	340	4.7	11.3	520	6.0

[1] Examinations performed in Finland were not included in the mean values for examinations of the pediatric trunk, since the consol DLP were not available (only the measured DLP_{16cm}).

Table 3: *Mean values of $CTDI_{vol}$ (mGy) and DLP (mGycm) displayed on the CT scanner consol together with the mean effective dose (E) (mSv) for examinations of the head, chest, abdomen and whole body reported from each Nordic country. In case of multiple scan sequences, the maximum $CTDI_{vol}$ and the total DLP for the examination are given. $CTDI_{vol}$ and DLP are related to 16 cm diameter phantom for head examinations and 32 cm diameter phantom for examinations of the trunk. Effective dose are calculated from DLP_{16cm}.*

HEAD	**NORWAY**			**SWEDEN**			**FINLAND**			**ICELAND**			**DENMARK**		
Age	**$CTDI_{vol}$**	**DLP**	**E**	**$CTDI_{vol}$**	**DLP**	**E**	**$CTDI_{vol}$**	**DLP**	**E**	**$CTDI_{vol}$**	**DLP**	**E**	**$CTDI_{vol}$**	**DLP**	**E**
0-1	18.6	206	2.6	28.4	653	6.1	26.6	381	7.2	40.2	424	5.9	21.0	305	-
1-2	24.6	273	1.9	27.5	443	3.4	30.7	504	3.9	44.8	850	4.1	22.0	289	-
2-5	29.3	334	1.5	33.7	625	2.9	23.3	437	2.1	68.9	980	2.8	29.6	415	-
5-10	48.4	512	1.8	41.6	611	2.5	44.4	627	2.3	49.4	624	2.6	43.5	696	-
10-16	38.0	647	2.3	53.6	756	2.7	47.4	740	2.6	76.0	129	1.7	62.5	627	-

CHEST	**NORWAY**			**SWEDEN**			**FINLAND[1]**		
Age	**$CTDI_{vol}$**	**DLP**	**E**	**$CTDI_{vol}$**	**DLP**	**E**	**$CTDI_{vol}$**	**DLP**	**E**
0-1	2.7	38	1.6	3.1	34	1.0	2.3	40	1.6
1-2	2.6	40	1.4	6.9	92	2.2	5.1	129	2.1
2-5	1.9	39	1.2	3.2	61	1.6	3.7	102	1.4
5-10	2.5	60	1.4	3.8	100	2.2	5.8	137	1.7
10-16	5.2	125	3.8	5.7	191	3.2	5.8	158	1.6

ABDOMEN	**NORWAY**			**SWEDEN**			**FINLAND[1]**		
Age	**$CTDI_{vol}$**	**DLP**	**E**	**$CTDI_{vol}$**	**DLP**	**E**	**$CTDI_{vol}$**	**DLP**	**E**
0-1	4.7	87	4.9	10.6	480	9.5			
1-2	4.6	122	5.8	4.8	141	4.3	3.2	261	6.1
2-5	3.1	98	3.1	4.7	177	5.4			
5-10	4.0	151	3.7	8.7	341	8.6	4.8	167	1.8
10-16	7.6	332	5.5	8.0	348	6.1	6.5	264	2.6

WHOLE BODY	**NORWAY**			**SWEDEN**			**FINLAND[1]**		
Age	**$CTDI_{vol}$**	**DLP**	**E**	**$CTDI_{vol}$**	**DLP**	**E**	**$CTDI_{vol}$**	**DLP**	**E**
2-5	19.4	498	7.7	4.1	156	2.7	2.5	130	2.0
5-10	20.1	561	6.9	9.0	516	9.2	2.7	162	1.8
10-16	18.8	472	7.6	9.5	569	7.2	5.6	373	3.3

[1] DLP values from Finland are expressed in DLP_{16cm} also for examinations of the trunk.

No attempt was done to further categorize the examinations after the clinical indication given in the questionnaires, because they were too vague and imprecise. Most common clinical indications were trauma, malignancy (including controls), infection and different respiratory disorders.

Large variations in the dose parameters within each scan area and age group were observed for every hospital. This is illustrated in Figure 1 for the applied $CTDI_{vol}$ in examinations of the head and abdomen. Despite this huge variation, an overall trend in reduction of $CTDI_{vol}$ with decreasing patient age was observed, with some exceptions for abdomen examinations of the youngest patients. A summary of trends in the variation of mean values of all the different dose parameters with respect to patient age are presented in Figure 2.

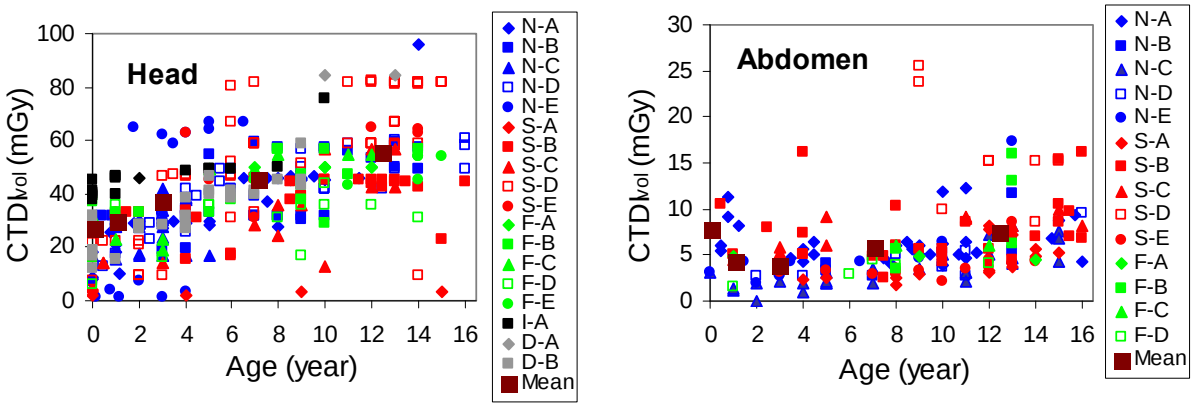
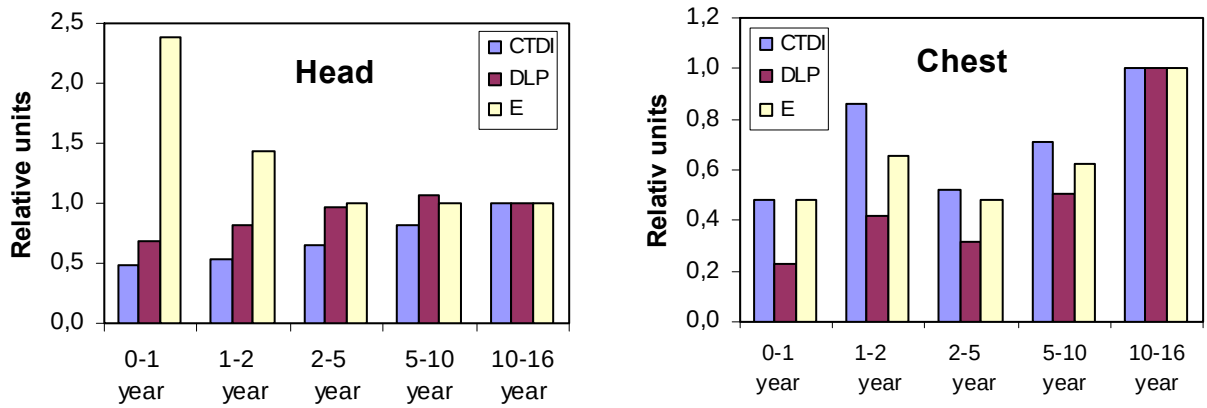


Figure 1: *$CTDI_{vol}$ from every single examination of the head and abdomen plotted against the patient age. N=Norway, S=Sweden, F=Finland, I=Iceland and D=Denmark. Also the mean $CTDI_{vol}$ from all the examinations is plotted against the mean age of all patients within each age group.*



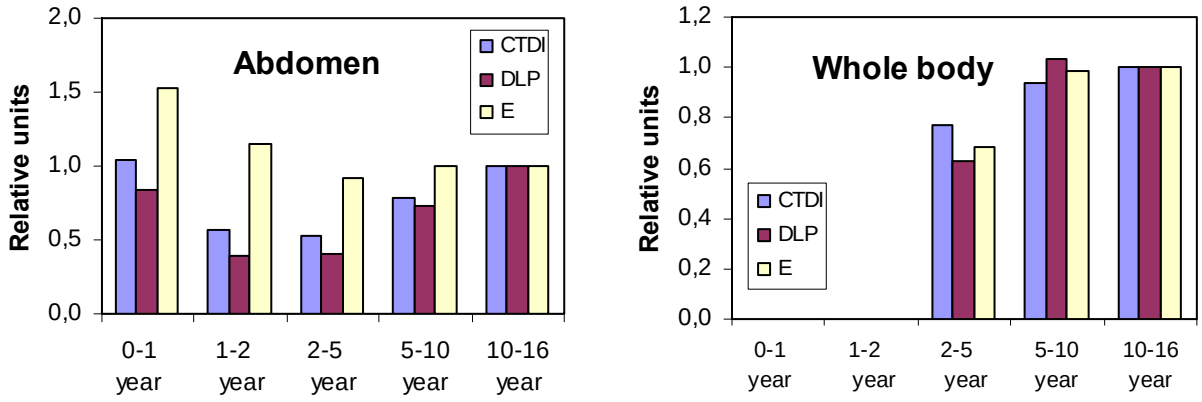


Figure 2: *Normalized mean values of $CTDI_{vol}$, DLP and effective dose (E) for CT examinations of the head, chest, abdomen and whole body from all Nordic countries. The normalization was made to the age group of 10-16 year to reveal variations in the dose parameters relative to this age group. DLP values from Finland are not included.*

A general trend of reduction (or maintenance) of the dose parameters were observed for decreasing patient age until reaching the age of two, for examinations of all four body regions. For the youngest patients, an increase in most of the dose parameters was observed, especially in the effective dose for examinations of the head and abdomen. Mentionable is the increase in effective dose despite of the decrease in $CTDI_{vol}$ and DLP for young patients undergoing head examinations. Typical min-max ratios between the mean dose values reported from each country ranged from 1.1 to 7.8 depending on scan area and age group.

To evaluate the level of optimization of the scan protocol according to patient size, applied mAs and kV were plotted as a function of patient age for examinations of the head and abdomen (Figure 3). The majority of the examinations were carried out at a tube voltage of 120 kV. Only some hospitals used a lower tube voltage for small patients and a higher one for large patients. On the other hand, an overall trend of reduced mAs with decreasing patient age was observed. The mAs-reduction factors for the different age groups obtained in this survey are summarized in Table 4. Typical numbers of subsequent sequences were 1.7, 1.1, 1.1 and 2.0 for examinations of respectively the head, chest, abdomen and whole body. Variations in the applied pitch and collimation were also observed among the hospitals, but not analyzed any further since these data were incorrect or insufficient filled out in the questionnaires. The observed reduction in mAs was the only indication of the presence of size-specific scan protocols at some hospitals in the Nordic countries.

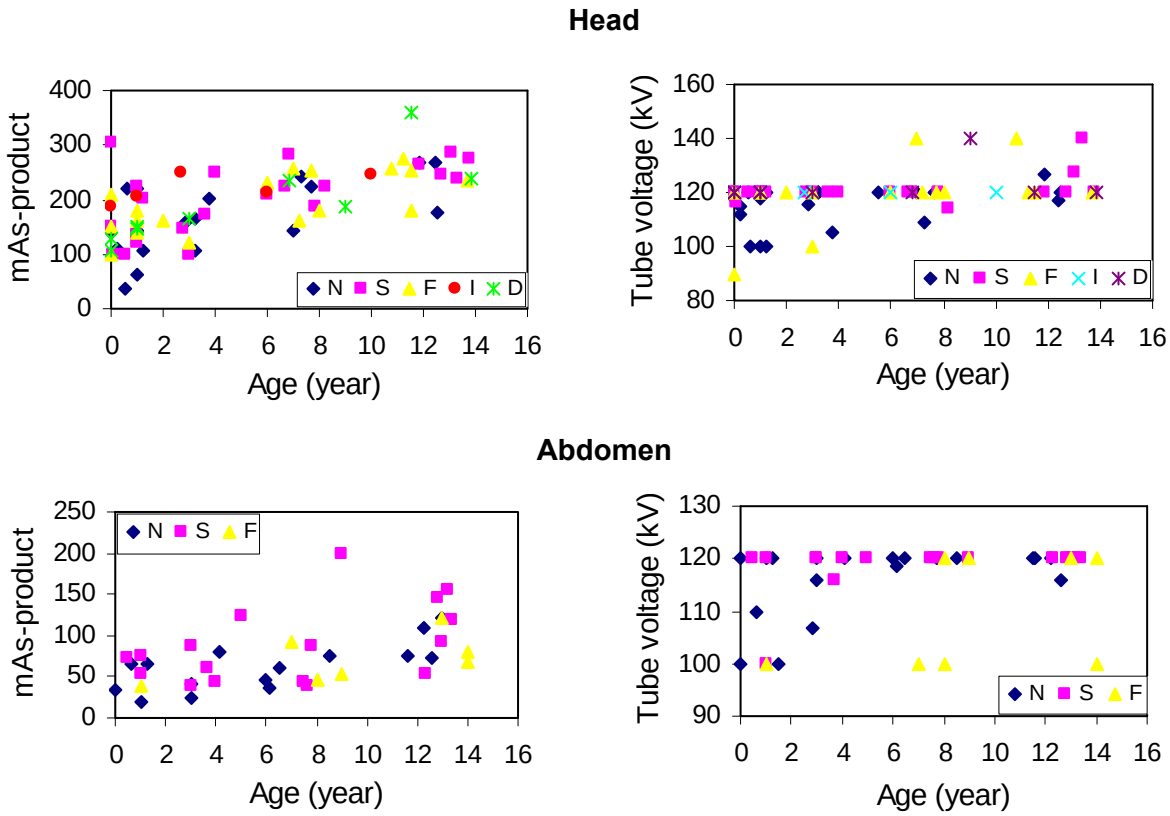


Figure 3: *Mean mAs-product and mean tube voltage as a function of the mean patient age within each age group for examinations of the head and abdomen from all 19 hospitals.*

Table 4: *Average mAs-reduction factors for each age group relative to the patient group of 10-16 years for examinations of the head, chest, abdomen and whole body. The reduction factors are based on the mean mAs-values from all Nordic countries.*

Age [year]	Head	Chest	Abdomen	Whole body
0-1	0.58	0.54	0.62	-
1-2	0.64	-	0.51	-
2-5	0.69	0.69	0.54	0.62
5-10	0.83	0.71	0.73	0.68
10-16	1.00	1.00	1.00	1.00

Discussion

Dose estimations from pediatric CT examinations are not an easy task and few dose surveys are carried out for pediatric patients. The main problem in dose estimations from pediatric CT is the large variation in patient size. It is demonstrated that the absorbed dose to dosimetric phantoms of different size is increasing with decreasing phantom diameter [6, 7]. It is therefore recommended that, irrespective of patient age and scan location, doses for all pediatric examinations should be expressed in terms of absorbed dose to the 16 cm diameter phantom [8]. Despite of this, most CT manufacturers still display the dose indicators ($CTDI_{vol}$ and DLP) related to the 32 cm diameter phantom for all examinations of the pediatric trunk. Generally, $CTDI_w$ measured in the 16 cm diameter dosimetry phantom is about twice

that for the 32 cm diameter phantom, under similar exposure conditions [5]. Therefore, consol $CTDI_{vol}$ and DLP do not reflect the physical absorbed dose to the patient, but still constitute a useful tool in dose surveys and optimization of examination scan protocols. To ensure reliable dose readings, it is important that validation of the consol $CTDI_{vol}$ and DLP is integrated in the routine quality control of the CT scanner performed locally at the hospital by the medical physicist.

The most common method for estimation of the effective dose is by use of DLP to effective dose conversion coefficients (E_{DLP}) for specific body regions. As already mentioned, the absorbed dose to the patient will increase with decreasing patient size. For effective dose estimates to pediatric patients, these "patient size-organ dose-effects" has to be included in the E_{DLP}. E_{DLP} expressed as a function of patient size (e.g. equivalent diameter), will probably give the most reliable dose estimate for pediatric patients. It is crucial that the DLP used in the dose estimation is related to the same dosimetry phantom as the DLP used for deriving the E_{DLP}. Most E_{DLP} for pediatric patients are based on DLP related to the 16 cm diameter phantom, and hence the consol DLP can not directly be used in effective dose estimations. If DLP_{16cm} is calculated from the scan parameters, the effect of additional rotations necessary for data interpolation at either side of the planned image volume in helical scanning (overscan) has to be included. DLP displayed on the scanner consol are assumed to include this effect.

Table 5 summarizes doses from this survey with doses obtained from a similar survey carried out in the United Kingdom (UK) [9]. Mean E_{DLP} used in the effective dose estimation for the different age groups are also presented. The Nordic $CTDI_{vol}$ values were 8-35% higher than those obtained in the UK survey for head examinations and 50-20% lower than the UK values for chest examinations. The observed differences in DLP values between the two surveys did not follow the variations in $CTDI_{vol}$, indicating variations in the number of sequences and scan length. The differences in DLP were most dominant for small patient. The observed variation in effective dose between the surveys reflects the differences in DLP and E_{DLP}. The variation in E_{DLP} was most dominant for examinations of 0-1 year old patients. Differences in patient age and clinical indications between the two surveys will also affect the dose values and participate to the observed variations.

Few signs that indicate the presence of size-specific scan protocols were found in this survey. Most significant was the overall trend of mAs-reduction with decreasing patient age observed in Figure 3. On the other hand, the same figure illustrate that mAs-adjustments are not a routine procedure for examinations of pediatric patients in Nordic hospitals. It was not possible to determine if this trend solely was a result of the automatic exposure control (AEC) used at some hospitals, or also represent the practice from hospitals not using AEC. When considering changes in kV, some hospitals reduced the kV for small patients, but the majority used 120 kV for all patient sizes. This way of optimization the scan protocols with respect to patient size (constant kV and mAs-reduction) are also reported by others [2, 10].

This survey revealed large variations in CT practice and a low level of optimized size-specific scan protocols for pediatric patients among the Nordic hospitals, indicating an urgent need for optimization. Optimization of scan protocols is based on the balance between a reduction of radiation dose to the patient and maintaining an image quality good enough to answer the diagnostic question. The process consist of adjusting the scan parameters according to the patient size, reducing the number of sequences and limiting the scan length to cover only the region necessary to answer the clinical question. The main parameters affecting the $CTDI_{vol}$ (representing the average dose in the irradiated slice) are mAs, kV,

pitch and collimation. In the optimization process of $CTDI_{vol}$, it is important to have knowledge on how changes in the different scan parameters will affect radiation dose and image quality. Changes in mAs will mainly affect image noise, while changes in kV will also affect the image contrast. Changes in pitch and collimation are even more complex, affecting spatial resolution and image artifacts. Unfortunately, there is nothing like a universal CT technique to be adopted, due to physical differences between scanners from different vendors (e.g. bow tie filters, focal spot to detector distance, detector efficiency, etc.). It is therefore important to have a qualified medical physicist in diagnostic radiology locally at the hospital to assist in radiation output measurements and optimization procedures.

Table 5: *Comparison of $CTDI_{vol}$ (mGy), total DLP (mGycm) and effective dose (mSv) from this Nordic survey (N) with those reported in the UK survey (UK). E_{DLP} (mSv/mGycm) used in the surveys are also included.*

	HEAD						CHEST[1]					
Age	**0-1**	**0-1**	**5**	**5-10**	**10**	**10-16**	**0-1**	**0-1**	**5**	**5-10**	**10**	**10-16**
(mean[2])				**(7)**		**(12)**				**(7.4)**		**(13)**
	UK	**N**	**UK**	**N**	**UK**	**N**	**UK**	**N**	**UK**	**N**	**UK**	**N**
E_{DLP}	0.011	0.016	0.0040	0.0038	0.0032	0.0035	0.039	0.032	0.018	0.013	0.013	0.013
$CTDI_{vol}$	25	27	34	46	44	56	11	5.4	11	8	14	11
DLP	230	394	383	614	508	580	159	76	198	198	303	316
E	2.5	5.5	1.5	2.1	1.6	2.9	6.3	1.4	3.6	1.8	3.9	2.9

[1]$CTDI_{vol}$ and DLP from the Nordic survey are roughly converted to DLP_{16cm} (i.e. increased by a factor of two).

[2]Mean age of the patients within each age group from the Nordic survey

Conclusion

This survey reflects the current pediatric CT practice in the Nordic countries. Large differences in scanning technique and the resulting effective doses were observed, addressing an urgent need for optimization with respect to patient size. The lack of size-specific scan protocols for pediatric patients give rise to concern, since pediatric patients are more radiosensitive than adults. Many Nordic hospitals can reduce their mAs-values without undue loss of diagnostic information, resulting in a significant doses reduction to the pediatric patient. Last but not least, the most effective way to reduce pediatric doses is to ensure that the CT examination is justified.

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The trend in Radioiodine therapy for benign thyroid diseases in Denmark 1980 to 2006

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Abstract

This report focuses on 131-Iodine in treating benign thyroid disease in Denmark. Since the late 1980s the trend has been an increase in therapy and it is not until the last 6 years that a decline can be seen. The report will present several explanations to this trend, presented by several endocrinologists in Denmark. Furthermore, radiation protection of the administrating-, caretaking personnel at the hospital, family members and to the general public will be discussed, all based on ICRPs report *Radiological Protection in Medicine (2007)*.

Introduction

In Denmark, the radioactivity law from 1953 is closely connected to the use of I-131 for the therapy of thyroid diseases. Expressed in GBq, the administration of I-131 has the largest extent among the open sources used today. In the last few years the number of patient doses, most of which were given to thyreotoxicosis patients, were about 800. After the Tjernobyl accident in 1986 the number of doses dropped during a short period. From 1987 to 1998 the number of doses went op with almost a factor of 3. The increase was alarming when it comes to the radiation protection and the environmental aspects. In relation to the increase of the administrated doses and the growing problem with the radiation protection of the staff, the use of new types of iodine capsules improved the radiation protection. The capsule resulted in a simplification of the administration of patient doses compared to the administration of the iodine liquid solution. But the iodine waste still remains in the urine of the treated patient. The iodine discharge into the sewage system has become measurable and because of this the coast of Denmark is under close surveillance.

During the last 5 to 6 years the number of administrated doses has been decreased from a factor 3 to a factor 2 relative the 1980´s. How can the progress in the number of doses be explained? Should action have been taken in the 1980`s in response to the rising trend in radioiodine therapy? For example: Improved information and guidelines handled to the patients post treatment or limit the exposure of the patient's relatives.

Method

According to the order *no 954 from 23 October 2000 § 36* the responsible physician at the nuclear medicine department is required every year in March to submit a compilation to SIS (National Institute of Radiation Protection) presenting the number and the average administrated activity for all the types of examinations and treatments performed during the previous calendar year. By using the information from the 20 hospitals in Denmark that have treated benign thyroid diseases from 1980 to 2006 it is possible to describe a radioiodine therapy trend.

To explain the radioiodine therapy trend of benign thyroid diseases, the results was presented to endocrinologists in Denmark and they have then given their explanations to the therapy trend *(1, 3, 4, 5, 6 and 8)*.

Result

From 1980 and onward the trend is rising, except for a small dip after the Tjernobyl accident in 1986. The rising trend of iodine radiotherapy gives measurable radioactivity in many sewage systems in Denmark and the problem in radiation protection of the general public has been discussed. In 1998 the trend stops rising and becomes stable for a couple of years until 2002 when the trend starts to incline. There are several reasons that could explain the trend from 1980 to 2006.

1980-2000; factors explaining the increase in the number of iodine therapies:

- During this time period, surgery was preformed in many hospitals in Denmark and there were only few surgeons with expertise competence. In hospitals without expertise competence, the surgeries performed often lead to complications. This problem lead to the iodine radio therapy becoming more and more applied as a treatment form for many types of benign thyroid diseases.
- The Tjernobyl accident (1986) had a negative affect of the patients' attitude towards radiation, resulting in more patients choosing surgery instead of radioiodine therapy during a couple of years.
- During the 1990s, the treatment with I-131 was shown to have good effects and fewer complications compared to surgery. This lead to increasing popularity of I-131 therapy as a treatment form.
- In the end of the 1990s the iodine therapy treatment was more used than today. But during the last 10 years, the ability to distinguish the patients that will have the best outcome of an iodine therapy treatment has improved. Furthermore, there has been a centralization of competences that have developed the surgery.
- In this period there has been a long waiting list for surgery that has lead to transferring of patients to I-131 treatment instead of surgery.

2000-2006; factors explaining the decrease in the number of iodine therapies:

- New rules from the Food Ministry that took effects in June 2001. According to these rules, the house salt that is produced in Denmark for the home market shall be iodine enriched to increase the iodine intake among the inhabitants in order to prevent goitre. This increased intake of iodine reduces the possibility to treat patients with the advanced goitre since the thyroid will be saturated and there will not be enough uptakes to get a satisfying result with radioiodine therapy. This leads to surgery of this type of goitre.
- In the last few years, the waiting list for surgery has been reduced to normal, explaining the drop in treatments with I-131.

The middle administered activity has been almost the same during the year (fig. 1b.) which indicates that there is no increased radiation load due to an increase in administrated activity.

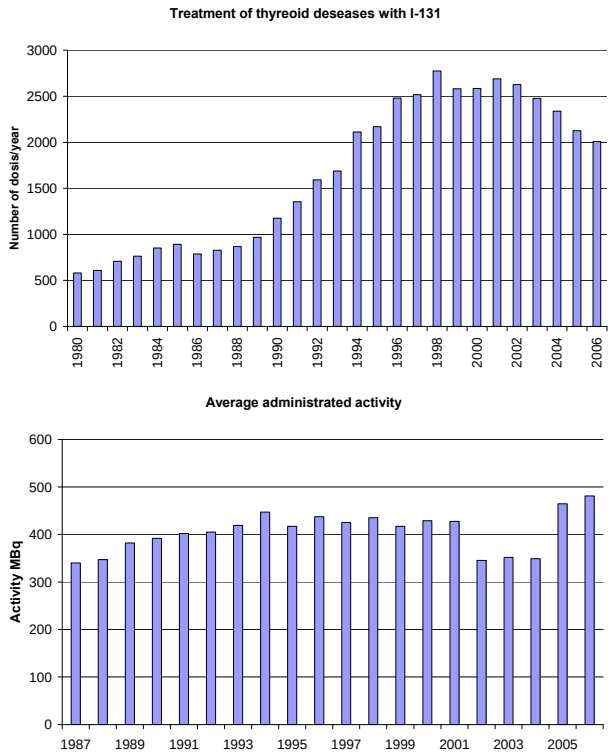

Figure 1.*a)Number of treatments of thyroid diseases b) Average administered activity.*

As seen in fig. 2, compared to Sweden, Denmark has a very high treatment frequency of benign thyroid disease which gives a higher amount of radioactive waste in the sewage system in Denmark compared to Sweden. There are probably many explanations for this. Two possible explanations: 1) there are differences between which types of goitre that are treated with I-131 in Sweden and in Denmark. In Denmark Graves´, thyrotoxicosis nodular goitre and non-toxic goitre are treated with I-131. 2) Sweden has added iodine to salt since 1966 and to milk cow's food since 1970 which could explain why the thyrotoxicosis nodular goitre and non-toxic goitre are much more common in Denmark than in Sweden (7). A similar effect is expected in Denmark, but it is probably too early to see the effects of the Danish iodine project (12) which started in 2001.

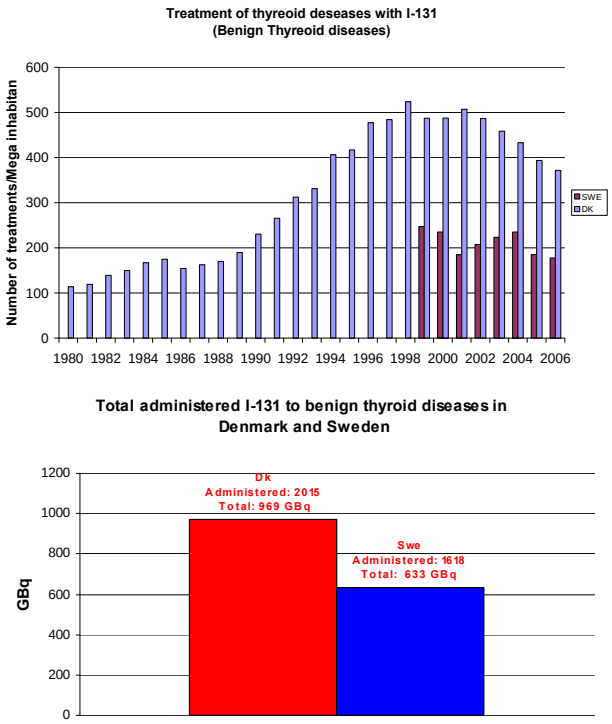

Figure 2 *Comparison of treatments numbers between Sweden and Denmark. There is not taken into account of which types of goiters they treat in respective country.*

a) Survey of how many therapies per mega inhabitants b) Total administered (GBq) in 2006

Discussion

In their report *Radiological Protection in Medicine 2007*, the ICRP discussed that dose constraints should be established for use in defining the protection policy for visitors and families at home. They also discuss that the system of protecting the staff from the source should be designated to minimize any sense of isolation experienced by the patient and that the publics´ access to controlled areas should be limited to patients´ visitors, who should be advised of any restrictions on their behaviors.

In the case of therapy with iodine-131 a study was made in Denmark: *Radiation dose to relatives of patients treated with iodine-131 on out –patient basis (Henrik Bertelsen og Klaus Ennow)*. The conclusion was that patients can be treated with activities up to 1800 MBq given as fractionated doses if special precautions are taken (family members < 60 years old should sleep separated for at least 7 days after the administration, family members > 60 years old don't need sleeping restrictions. If there are children in the household the administrated dose must not exceed 600 MBq). In other words it is important to make an individual written guidance to the patient and also discuss it with the patient and other people it may concern. In the order *nr 954 from 23 October 2000 appendix, point 4* it is described how the guidance should be formulated.

The iodine capsules have improved the administration of iodine to the patient. The handling has become *faster*, shortened exposure of hospital staff due to less preparing element, *safer*, less risk for contamination, and there is less radioactive waste. The dose to the personnel has become lower with the capsule compared to the use of drinkable iodine solution, earlier there could be measurable results on the personal dose meter, but today there are almost nothing.

In respect to the residual activity in patients undergoing radiotherapy, ICRP indicates that hospitalisations will decrease the exposure to the public but it will at the same time be a psychological

burden for the patient and an increased exposure of the hospital staff. It is the hospitals own decision to take! But it shall be written in the guidance and discussed with the patient before leaving the hospital, how he/she shall be acting among other people. Patients travelling after radiotherapy rarely present a hazard to other passengers if travel times are limited to a few hours.

The radioactive waste from the radioiodine therapy (iodine leftovers and patient urine) released into modern sewage system can be detected in the environment after medical uses and could also contribute with a small dose to the sewer workers and the public, but it is well below the public dose limits.

Conclusion

There are no imminent risks that there should be an enhanced dose load to the hospital staff, sewage workers and to the public. The therapy trend is decreasing and there are no indications that it will stop for the next few years. Perhaps it will decrease to the same level as in Sweden when the Danish iodine project (12) starts to give result. The radiation protection of the hospital staff has been improved due to the introducing of iodine capsule which have simplified the treatment of the patient.

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Dose assessments for interventional radiologists

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Abstract

Background

Interventional radiologists may receive significant radiation doses. The patient itself and the skill and awareness of the radiologist are important factors influencing the dose to the radiologist.

Purpose

The purpose of this study is to assess whether there are representative relations between patient dose, expressed in terms of Dose-Area-Product (DAP) or fluoroscopy time, and effective dose to the radiologist.

Material and Methods

By studying variations in dose to radiologists performing interventional procedures, we have assessed relations between radiologists' dosemeter readings and patient dose values like DAP and fluoroscopy times. Both effective dose to radiologists and their finger doses have been assessed. The accumulated two months dosemeter readings of six radiologists in different hospitals were also measured together with registrations of the total patient DAP values and fluoroscopy times used in their procedures.

Results

During individual procedures, there are large variations in patient doses and in doses to the radiologist, and the correlation between the patient dose (or fluoroscopy time) and radiologists dose is not very significant. For aggregated values over time (e.g. two month), however, the correlations are significant and some typical relations are suggested for busy radiologists performing a mixture of procedures over time.

Conclusions

Averaged over time, there is a good correlation between radiologist dosemeter reading and DAP and this relation could be representative for Norwegian hospitals. This relation seems to be in the same range as found in some other European countries.

Introduction

Some interventional radiology procedures require a large number of images and extended fluoroscopy times, and occupational exposure to interventional radiologists is of significant concern. [1-7]. The type and complexity of the procedure, the X-ray equipment and the use of it, the use of protective devices, and, not least, the experience and skill of the operator, are important factors influencing the dose to the radiologist.

Several studies, assessing occupational exposure, have been performed in different countries to investigate the influence of different factors such as type of radiological procedure, total Dose-Area

Product (DAP), fluoroscopy times, number of images and position of the radiologist. These investigations show large variations between individual procedures and between different countries and individual hospitals[1-7].

Material and method

In the present study, interventional radiologists from five major hospitals in Norway participated. In one of the hospitals, integrated doses to the personal dosemeter site at collar position outside the apron of one radiologist were measured by using the EDD-30 instrument (Unfors, Sweden). During all the procedures, the following information was recorded in addition to the badge site dose: Type of procedure, total DAP, fluoroscopy time, tube voltage.

Eight radiologists at three different hospitals were asked to use two routine dosemeters during a period of two months, which is the normally monitoring period. The radiologists were instructed to wear one of the dosemeters over the apron and one under, both at collar position. They were also asked to note the total DAP value for all the procedures they performed during the two-month period. The dosemeters used were those used routinely by the Norwegian Radiation Protection Authority, NRPA.

Finger doses were recorded by using the thermoluminescence dosemeters (TLD) DXT_RAD Extremity dosemeters.

Results

In Fig.1, the dose at dosemeter site is given vs. DAP for individual procedures, and in Fig. 2 the results are given vs. fluoroscopy time. In Fig. 3, accumulated badge doses above the protective apron are given for a two-month period. In Fig 4 and Fig. 5, the results of the finger measurements are shown.

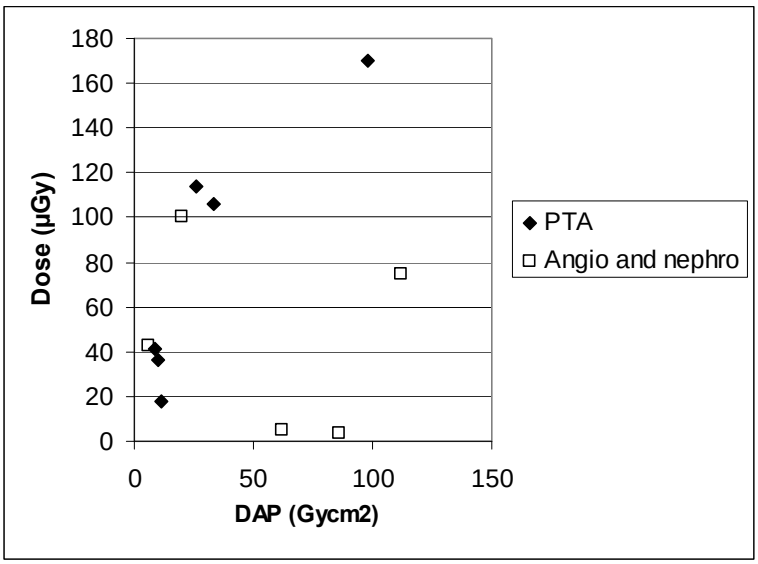

Figure 1. *Dose above lead apron vs. DAP for individual procedures*

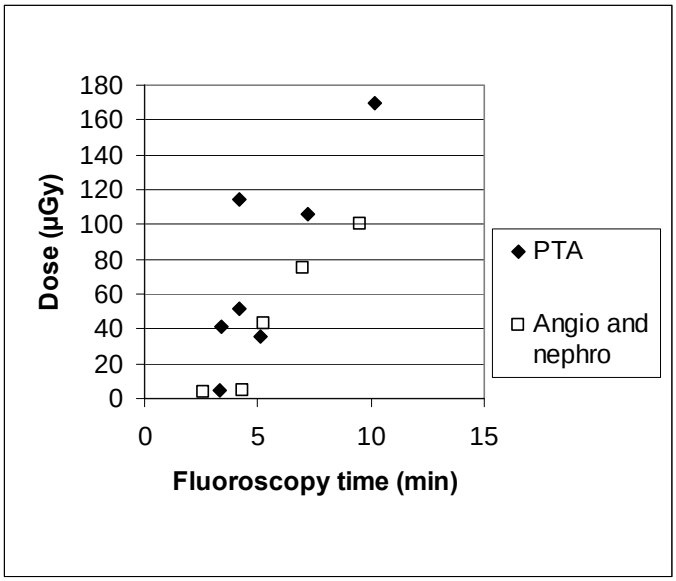

Figure 2. *Dose vs. fluoroscopy time for individual procedures*

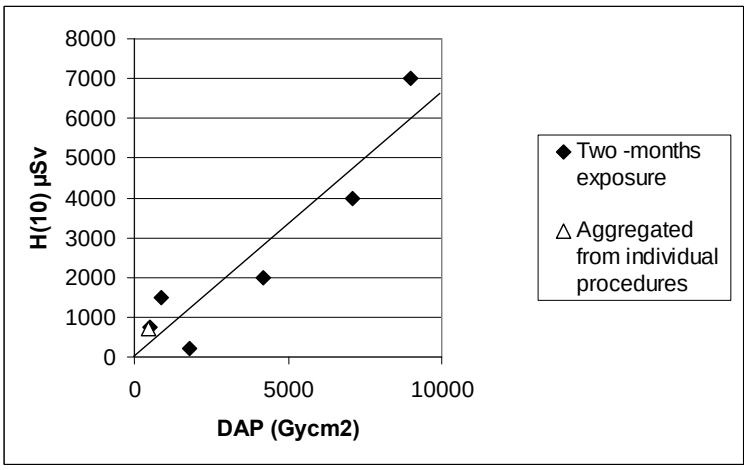

Figure 3. *Aggregatet dosemeter dose vs. DAP*

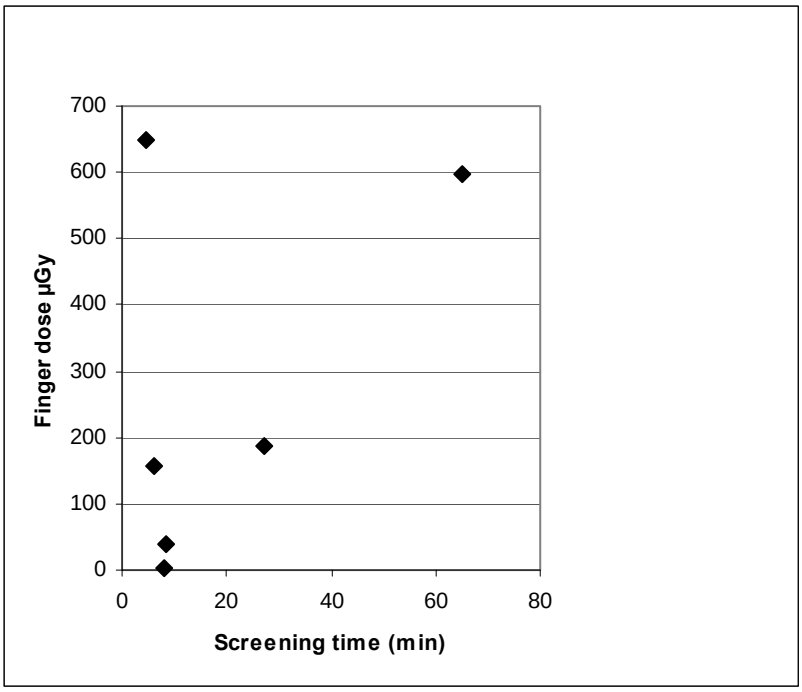

Figure 4. *Finger dose vs. fluoroscopy time for individual procedures*

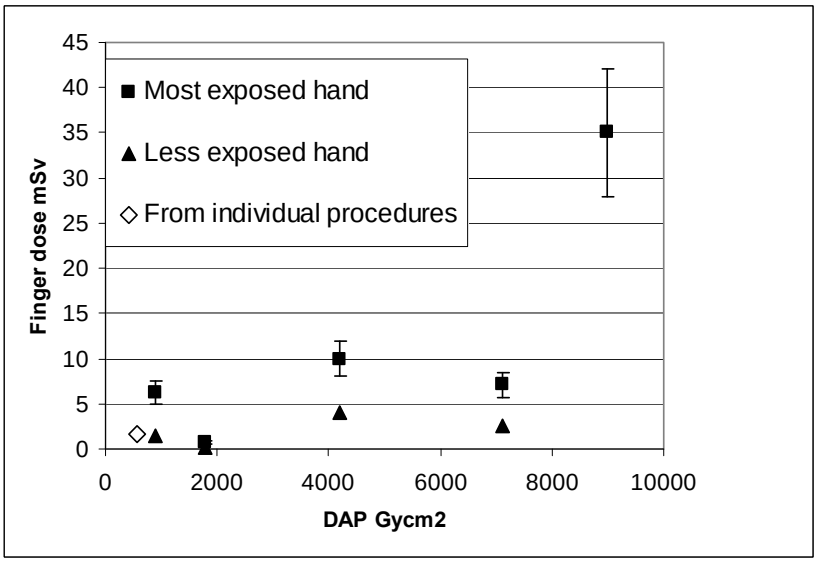

Figure 5. *Aggregated finger doses vs. DAP*

Discussion

The correlation between DAP and dose to radiologist is low for individual procedures, but aggregated over time the correlation is good. The aggregated badge dose per unit DAP is found to be about 0.7µSv per Gycm^2 .By using protective apron and thyroid shield, the effective dose has been assessed to about 2 % of the badge dose H(10). From this study, effective dose per DAP could be estimated to 10-20 nSv per Gycm^2.

The accumulated dose to the most exposed hand correlates well with the accumulated DAP value, and also with H(10) measured above the apron. The dose to the fingers of the most exposed hand is

typically hundred times larger than the effective dose, when protective apron and thyroid shield is used.

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Linsedoser til personale ved intervensjonsprosedyrer

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Bakgrunn

I intervensjonsradiologi kan stråledosene til personale bli signifikante. Dosen fra en gitt prosedyre avhenger blant annet av type prosedyre, arbeidsteknikk og type apparat. Et persondosimeter båret utenpå blyfrakk gir en indikasjon på strålebelastningen, men i tillegg kan det være nyttig og se på doser til fingrene og øyelinsene. Denne rapporten presenterer en undersøkelse av linsedoser ved ulike intervensjonsprosedyrer og hvordan disse avhenger av ulike faktorer.

Metode

Prosedyrene ble utført med en Siemens Multistar t.o.p, underbordsrør. Fem radiologer gjennomførte prosedyrene. Øyelinsedoser ble målt med et diodeinstrument (EDD, Unfors) som ble festet på brillene til radiologen på siden nærmest bildeforsterker.

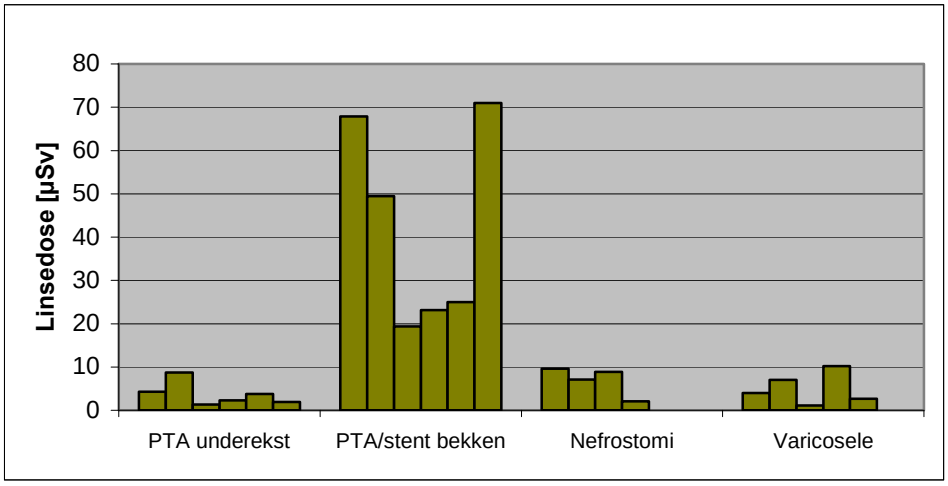
Prosedyrene ble delt opp i fire kategorier: PTA (percutaneous transluminal angioplasty) underekstremiteter, PTA/stenter bekken, nefrostomi og varicosele. Ved PTA brukes en ballong til å åpne opp tette årer og stenter kan føres inn for å holde årene åpne. Nefrostomi gjøres for å få god drenasje fra nyrer når denne funksjonen er nedsatt. Ved varicosele behandler man forstørrede og utvidede vener rett over skrotum.

Dosearealprodukt (DAP) ble registrert ved alle prosedyrene, og andelen av total DAP som skyldtes gjennomlysning og hvor stor del som skyldtes eksponeringer ble også registrert.

Resultater

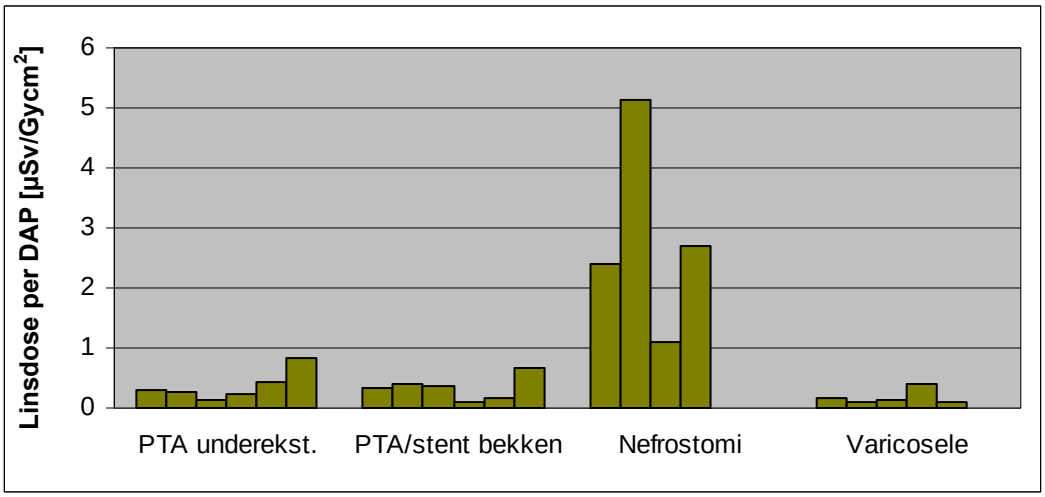
Ved PTA og varicosele ligger andelen av DAP fra eksponeringer mellom 70-99 %. Ved nefrostomi varierer det mellom 0-80 %.

Figur 1 viser linsedosen målt i hver undersøkelse. Linsedosen er høyest ved PTA/stent bekken-prosedyrer.



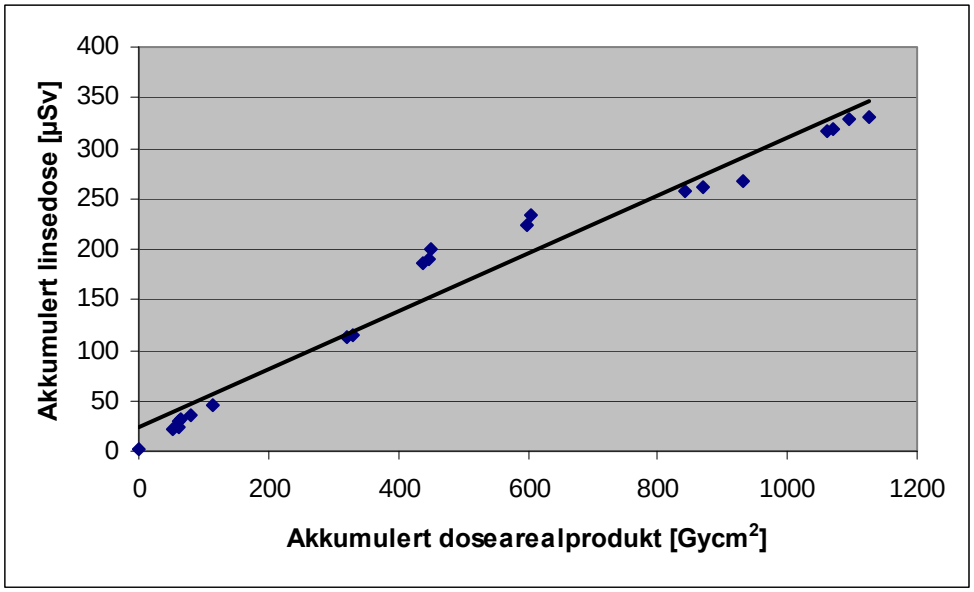
Figur 1. *Linsedose målt i µSv per undersøkelse. I gruppen PTA underekstremitet og PTA/stent bekken målte vi på seks undersøkelser, i nefrostomi målte vi på fire undersøkelser og i varicosele målte vi på fem undersøkelser.*

Linsedose per DAP for hver undersøkelse er vist i figur 2. Nefrostomi gir høyere linsedose per DAP enn de andre undersøkelsene.



Figur 2. *Linsedose per DAP i µSv/Gycm^2 for undersøkelsene vist i figur 1.*

Figur 3 viser sammenhengen mellom akkumulert linsedose og akkumulert DAP over alle undersøkelsene uavhengig av prosedyre. Stigningen på grafen er 0,3 µSv/Gycm^2.



Figur 3. *Akkumulert linsedose [µSv] mot akkumulert dosearealprodukt [Gycm^2] for alle undersøkelser uavhengig av type prosedyre.*

Diskusjon/konklusjon

Linsedosen varierer mye mellom prosedyrer men også innenfor samme type prosedyre[1-6]. Vanskelighetsgraden, og dermed gjennomlysningstiden og dose, varierer fra pasient til pasient og pasientstørrelse har stor innvirkning på linsedosen.

Andelen av DAP som kommer fra eksponeringer lå ved PTA og varicosele mellom 70 og 99 %. Dette betyr at mye av linsedosen til radiolog kan reduseres ved å trekke seg litt tilbake når bildene taes. Ved nefrostomi taes det få bilder og gjennomlysningstidene er korte slik at andelen av DAP som kommer fra eksponeringer varierer veldig.

PTA/stenter bekken gir høyeste linsedose til personale av de prosedyrene vi så på. Dette er fordi undersøkelsen gjøres i et område med mye bein slik at automatikken på gjennomlysningsapparatet kjører opp doseraten for å beholde bildekvaliteten. Samtidig kan disse prosedyrene være vanskelige og tidkrevende i forhold til nefrostomi og varicosele.

Når man ser på linsedose per DAP ligger nefrostomi høyere enn de andre undersøkelsene. Dette kan være fordi radiologen står nært inntil røntgenrøret og det er ikke alltid plass til en blyskjerm i mellom. I tillegg er det en veldig rask prosedyre med lite stråling slik at blyskjermen ikke blir sett på som like viktig som ved en tyngre prosedyre.

Noe av variasjonen i linsedose jevnes ut når man ser på akkumulerte verdier over tid. Det er derfor mulig å anslå hvordan årsdosen til øyelinsen i snitt henger sammen med det totale antall prosedyrer en radiolog gjennomfører i løpet av et år.

Fig.3 viser det lineære forholdet mellom akkumulert linsedose og akkumulert DAP (korrelasjonsfaktor 0,98). Dette betyr at for denne laben vil øyelinsedosen til radiolog (uten bruk av blybriller) være samlet DAP multiplisert med faktoren 0,3 μSv/Gycm^2. Dette stemmer godt med tall funnet tidligere[6].

Takk til

Radiologene og radiografene på lab 5, Radiologisk avdeling, Aker universtitetssykehus.

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Linsedose til personell under ERCP - prosedyrer

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Bakgrunn

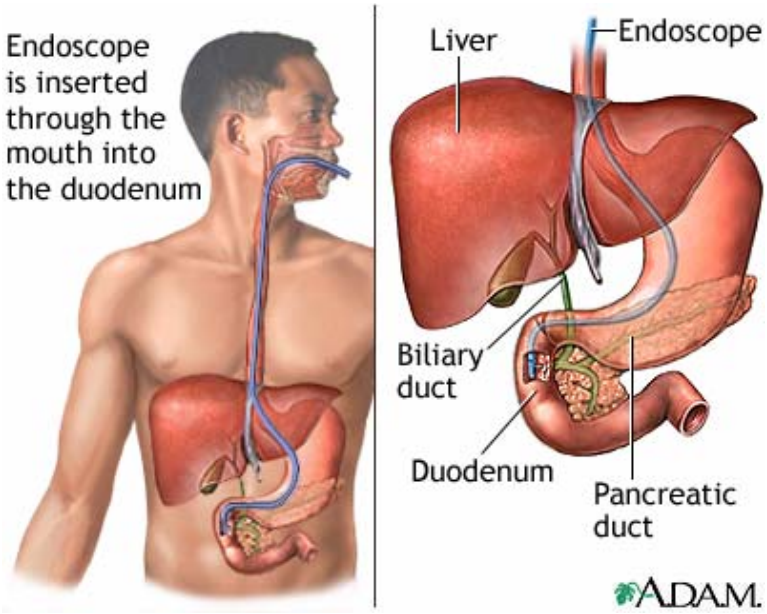
Ved et sykehus ble personell fra gastrokirurgisk avdeling tildelt en røntgenlab, uten muligheter for pulset gjennomlysning, for ERCP - prosedyrer. Det er et eldre apparat, og systemet har overbordsrør. Strålevernskontakten på radiologisk avdeling uttrykte bekymring for dosen til personalet, både på grunn av systemets geometri, og ikke - radiologisk personells kunnskap til systemet og strålehygiene.

Prosjektet ble satt i gang for å kartlegge linsedosen personalet ble utsatt for, da det er kjent at ERCP - prosedyrer innebærer en del gjennomlysning og flere eksponeringer. Dette vil si at det er en fare for at pasienten kan få relativt høye doser, og dermed gi mye spredt stråling.

Dosene til personalet ble sammenlignet med dose fra en annen lab med underbordsrør, fra et annet sykehus.

Metode

ERCP (Endoscopic Retrograde CholangioPancreatography) er en metode for kontrastframstilling av intra- og ekstrahepatiske galleveier samt pankreasgang. Indikasjonene for prosedyren er sterk mistanke om sykdom i galleveier eller pankreas, der CT eller MR ikke har gitt noe avklaring. En annen indikasjon kan være behov for terapeutiske prosedyrer (steinfjerning, papillotomi eller stenting). Det brukes gjennomlysning for å se på plasseringen av endoskopet, og for å se etter tilstoppelser i gallegang ved hjelp av innsprøytet kontrast. Figur 1 viser plassering av endoskop nær gallegangene.



Figur 1 *ERCP fra www.nlm.nih.gov*

Det ble gjort målinger under ERCP – prosedyrer på røntgenlab ved to forskjellige sykehus. Det ene sykehuset kjører ERCP på en lab med systemet Prestige VH, fra GE, som er et system med overbordsrør. Det andre sykehuset kjører prosedyren på en lab med underbordsrør, Polystar fra Siemens.

Gjennomlysningsparametrene ved ERCP prosedyrene på Prestige VH varierte fra 66 – 91 kV, og 3,09 - 6,33 mA, hvor gjennomsnittlig gjennomlysningstid var 7,9 minutter, og det ble tatt fra 4 til 15 eksponeringer per prosedyre. Systemet beregner ESD (cGy) til pasienten i 70 cm fra fokus.

Det ble ikke registrert mA og kV på laben med underbordsrør, men gjennomsnittlig gjennomlysningstid var 7,3 minutter, og det ble tatt fra 4 til 13 eksponeringer per prosedyre. Dette tilsvarer gjennomlysningstid og antall bilder fra det første sykehuset. Systemet registrerer DAP ($cGycm^2$) til pasienten.

Det var flere forskjellige leger som utførte prosedyren ved begge sykehusene. Variasjon i erfaring, og komplikasjoner ved prosedyren kan gi utslag i gjennomlysningstiden, og antall bilder.

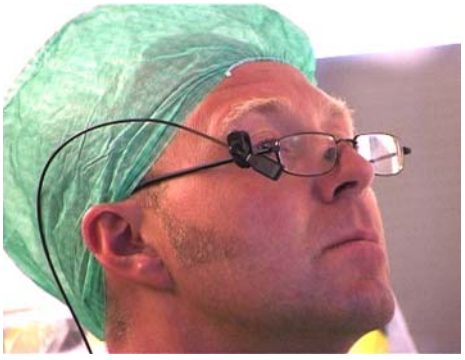
En elektronisk dosemåler, Unfors EDD 30, Educational Direct Dosemeter, Unfors Instruments AB, Billdal, Sverige, ble festet ved det øye som var nærmest pasienten. Detektoren ble enten limt med tape til tinningen, eller festet til brillestangen, figur 2.

Det er to sykepleiere og en lege på lab under prosedyren.

Den ene sykepleieren assisterer legen med endoskopet, den andre står ved pasientens hode.

I motsetning til legen, som står med samme side mot pasienten, beveger sykepleierne seg mye rundt på laben. De står også lenger unna, noe som medvirker til at dosen til sykepleierne ikke blir så høy som legens.

Personalet som ble målt, hadde dosemåleren på under hele prosedyren.



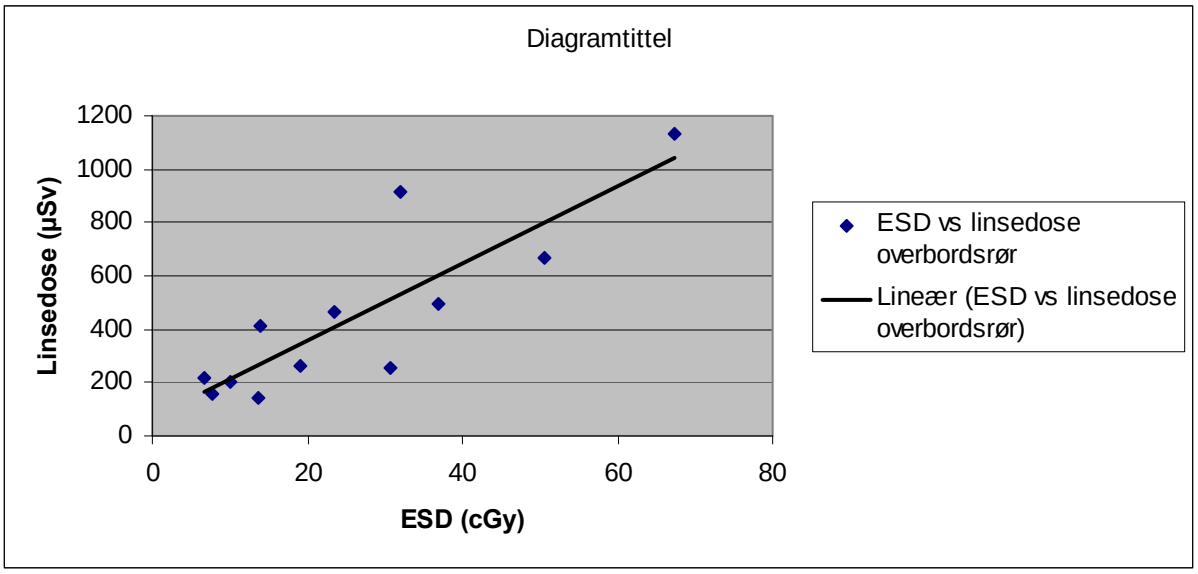
Det ble gjort 57 målinger totalt i prosjektet, og etter endt prosedyre ble linsedose, gjennomlysningstid, utførende lege, antall bilder og total ESD/DAP registrert.

Figur 2 *EDD detektoren fra Unfors, og plassering av denne (www.unfors.se)*

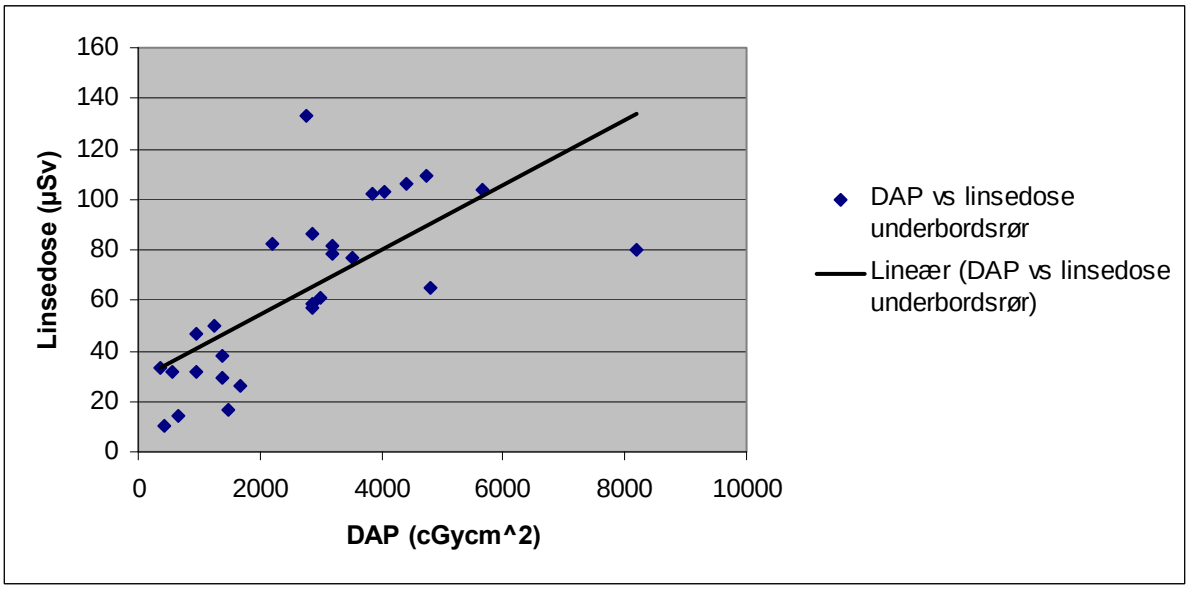
Resultatene fra målingene på sykepleierne blir sett bort fra i disse analysene. Det er kun målingene gjort på legene, som analyseres i disse resultatene.

Resultat

Resultatet fra målingene viser en lineær sammenheng mellom dose til pasient og linsedose til personell. Og følgelig også en lineær sammenheng mellom linsedose til personell og gjennomlysningstid.

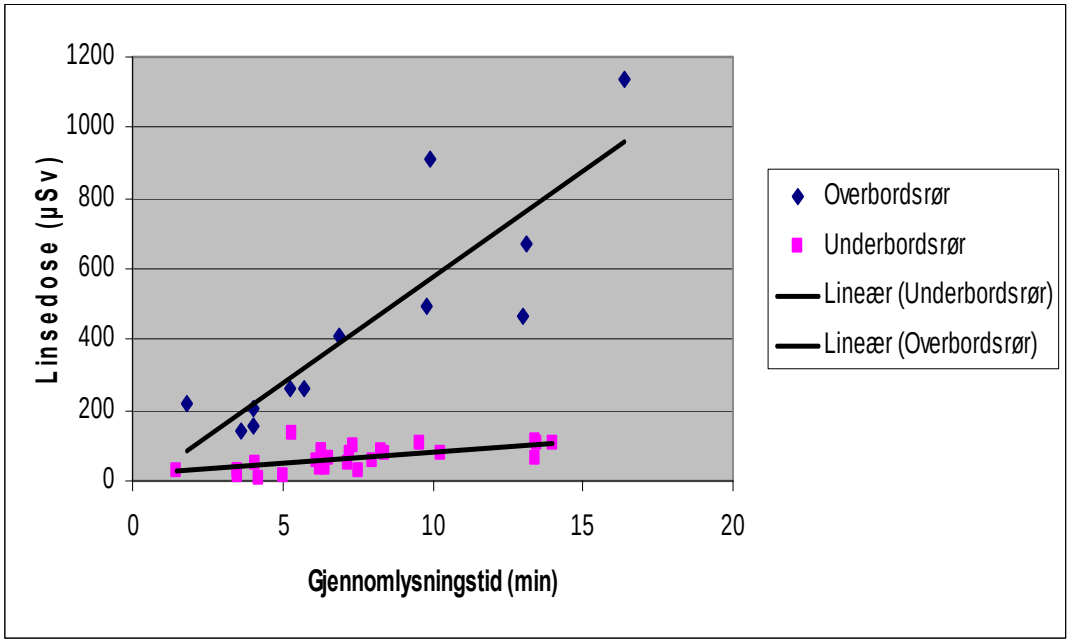


Figur 3 *Linsedose vs ESD ved bruk av overbordsrør*



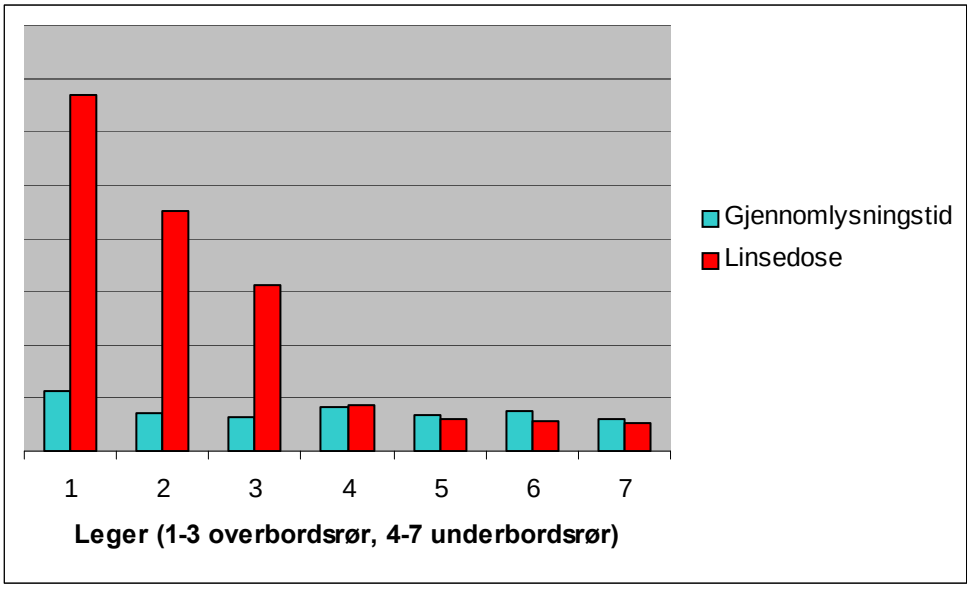
Figur 4 *Linsedose vs DAP ved bruk av underbordsrør*

Figur 3 viser sammenhengen mellom ESD og linsedose til gastrokirurgen på lab med overbordsrør, og figur 4 viser sammenhengen mellom DAP og linsedose ved bruk av underbordsrør.



Figur 5 *Linsedose vs gjennomlysningstid ved over- og underbordsrør*

Figur 5 viser linsedose til lege ved bruk av både over - og underbordsrør.



Figur 6 *Linsedose og gjennomlysningstid.*

Figur 6 viser variasjon i linsedose mellom forskjellige leger. Lege 1, 2 og 3 har system med overbordsrør, og 4, 5, 6 og 7 har system med underbordsrør.

Diskusjon og konklusjon

Variasjonene i målingene avhenger av vanskelighetsgrad på undersøkelsen, og legenes erfaring. Det ble derfor gjort flere målinger på forskjellige leger. I snitt er det likevel lite avvik i gjennomsnittlig gjennomlysningstid. Det er tydelig at overbordsrør gir betydelig høyere linsedose til gastrokirurgen i løpet av en ERCP prosedyre.

Målingene viser at ved overbordsrør er forholdet mellom linsedose og ESD 18,1 mSv/cGy, og ved underbordsrør er forholdet mellom linsedose og DAP 2,3 mSv/Gycm^2.

Fra disse målingene kan en beregne at gjennomsnittlig linsedose per minutt er 52,8 µSv for overbordsrør. På det ene sykehuset ble det oppgitt at en gastrokirurg gjør 6 ERCP'er i uken, 47 uker i året, det vil si at legen kan få nærmere 120 mSv til linsen. Med andre ord vil 8 gjennomsnittsprosedyrer i uken gi mer enn 150 mSv/år (anbefalt dosegrense til linsen, ICRP), og for underbordsrør må man til sammenligning opp i 51 gjennomsnittsprosedyrer i uken, for å passere 150 mSv/år.

Konklusjonen er at ERCP prosedyrer bør ikke gjøres på lab med overbordsrør.

Takk til

Alle som har bidratt i prosjektet fra Aker Universitetssykehus, Akershus Universitetssykehus og Ullevål Universitetssykehus.

Strålevern ved Norges første PET-senter, skjerming og sikkerhetssystem

Berit Brosvik

Norsk medisinsk syklotronsenter AS

Abstrakt

Norges første PET-senter ved Rikshospitalet HF ble åpnet 24. oktober 2006. PET-senteret skal være en nasjonal ressurs for kliniske undersøkelser og forskning. PET (positron-emisjons-tomografi) er en undersøkelsesmetode som gjør det mulig å måle detaljert fordeling i kroppen av et radioaktivt stoff som en har injisert intravenøst. Radioaktivitet kan bygges inn i et stort antall ulike forbindelser – såkalte biomolekyler – til bruk for PET. Ved PET-senteret er det i dag etablert rutinemessig produksjon av [^{18}F]FDG, en radioaktiv variant av druesukker.

Felles for PET nuklidene er kort halveringstid og utsendelse av positroner som meget hurtig annihilerer og sender ut fotoner med en energi på 511 keV. På grunn av den korte halveringstiden må legemiddelet produseres samme dag som det skal benyttes og med høy radioaktivitet. Dette krever god skjerming og en rekke sikkerhetssystemer i produksjonslokalene. Daglig blir det produsert ca 130 GBq ^{18}F ved syklotronen. Syklotronen som ved produksjon avgir gamma og nøytronstråling i et vidt energispekter, er skjermet bak 1,7 meter tykke betongvegger. [^{18}F]FDG blir videre syntetisert og dispensert i hotceller. Sikkerhetssystemer som interlock på dør til syklotronhvelvet, til hotcellene og på overføringssystem fra syklotron til hotceller er installert. Strålenivået i produksjonsområdet samt utslipp blir kontinuerlig overvåket.

Det er gjort en vurdering av skjermingsbehov ved prosjektering av PET-senteret og en rekke målinger etter oppstart for å kartlegge om skjermingen er tilstrekkelig. Målingene har i noen tilfeller avdekket behov for tilleggskjerming av hotceller og syklotronhvelv.

Sikkerhetssystemer og skjerming som er installert ved PET-senteret i Oslo skal sammen med gode arbeidsprosedyrer bidra til lave persondoser til personale og til allmenn befolkning.

Etablering av PET-senteret

Norges første PET-senter er etablert i et nytt bygg i tilknytning til Rikshospitalet HF på Gaustad i Oslo. Norsk medisinsk syklotronsenter AS (NMS AS) har investert til sammen 128 mill kroner i bygg og utstyr for produksjon av PET-legemidler og to PET/CT-skannere, hvorav en er plassert på PET-senteret og en på Radiumhospitalet. Dette er den største enkeltsatsingen på avansert medisinsk forskning og diagnostikk som er gjennomført i Norge, og er kommet i stand takket være et spleiselag mellom GE Healthcare AS, Norges Forskningsråd og Kunnskapsdepartementet.

NMS AS leier ut deler av PET-senteret til GE Healthcare for produksjon av legemiddelet [^{18}F]FDG og Rikshospitalet avd. Nukleærmedisin for kliniske undersøkelser ved PET-senteret. Norsk medisinsk syklotronsenter AS eies av universitetssykehusene i Oslo området og Universitetet i Oslo. Selskapet er opprettet for å muliggjøre bruk av PET i medisinsk forskning og diagnostikk.

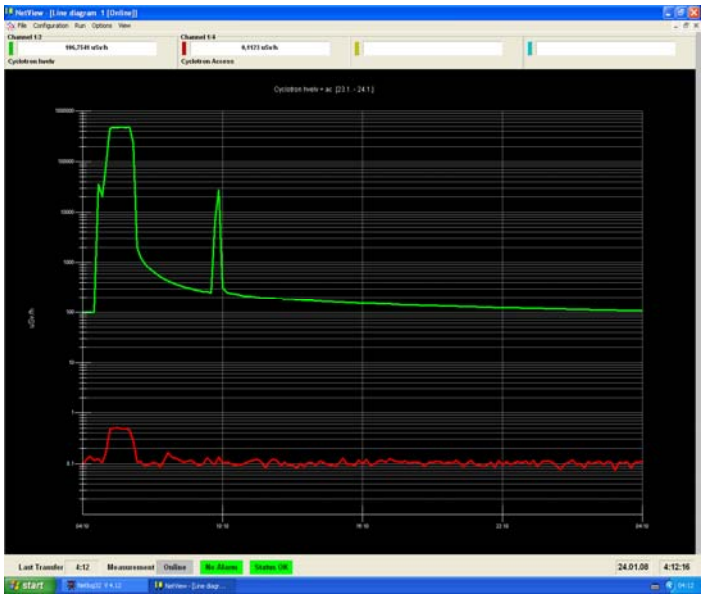
Hva er PET?

PET (Positron Emisjons Tomografi) er en metode brukt for diagnostisering innen onkologi, nevrologi og kardiologi. Radioaktive atomer kan hektes på en rekke ulike forbindelser (biomolekyler), dette er legemiddelet (radiofarmaka) som blir sprøytet inn i kroppen til pasienten. Ved PET senteret benyttes det foreløpig kun [^{18}F]FDG (^{18}F-fluordeoksyglykose), eller en variant av druesukker, for pasientundersøkelser. Dette sukkeret vil bli tatt opp i celler med høyt energi forbruk, som typiske kreftceller, hjerne og hjerte. Etter at sukkeret har fått tid til å fordele seg i kroppen blir pasienten skannet i PET/CT-skanner, og områder der det radioaktive stoffet har samlet seg vil lyse opp på bilde.

Strålevernshensyn ved PET-senteret

PET nuklidene har typisk kort halveringstid, mindre enn 2 timer. Legemiddelet må derfor produseres samme dag som det skal benyttes og med høy start aktivitet. Ved PET-senteret produseres det daglig ca 130 GBq ^{18}F ved syklotronen, som er av typen GE PETtrace 6. Denne akselererer negativt ladede hydrogenioner sirkulært vha vekselstrøm og kraftig magnetfelt. Denne type syklotron er utviklet for produksjon til PET-legemidler. Partiklene akselereres til 16,4 MeV protoner eller 8,4 MeV deutroner, og styres til bestrålingskammeret (target). For produksjon av ^{18}F bestråles ^{18}O vann og vi får reaksjonen ^{18}O (p,n) → ^{18}F. PET nuklidene eller positronene annihilerer og sender ut to fotoner med energi 511 keV i motsatt retning. Områdene rundt syklotronen må skjermes for (p, n) reaksjoner og ved all håndtering av ^{18}F må en ta hensyn til gammastråling med energien 511 keV.

Strålenivået ved syklotronen ved stråle på target (”beam on”) er anslått til å være mellom 5 og 10 Sv/t i nærheten av target. Protonstrålen og nøytronstråling aktiverer materialer i syklotronhvelvet, hovedsakelig i nærheten av bestrålingskammer hvor energien på strålingen er høyest. Denne aktiveringen fører til at selv om syklotronen er slått av har vi stråling i syklotronhvelvet, noe som må tas hensyn til ved vedlikehold på syklotronsystemet. Figur 1 viser hvordan strålenivået i hvelvet ved ^{18}F produksjon avtar på et døgn.



Figur 1 *Strålenivå i syklotronhvelv*

Figur 1 *er hentet fra radiologisk overvåkingsanlegg og viser strålenivå i Syklotron Hvelv (grønn kurve) og i Syklotron Access (rød kurve) i løpet av et døgn. Toppene på grønn kurve viser ”beam on” ved Syklotron. Strålenivået i syklotronhvelvet er over måleområdet til detektoren ved stråle på target.*

^{18}F som er frie F^- ioner i vann overføres til hotcelle i produksjonslab (Cleanroom) via blyskjermede kanaler i gulv. Etter syntese og dispensering i hotcelle tas produktet [^{18}F]FDG ut fra hotcellen i wolframbeholdere for transport, operatørene er dermed ikke direkte i kontakt med produktet. Risiko for kontaminering kan oppstå ved uhell der hetteglass knuses eller ved renhold av utstyr brukt i tilknytning til produksjonsutstyr. F^- ioner er meget reaktive, det kan derfor være vanskelig å få vasket bort ev. kontaminering.

Rikshospitalet HF Avd. Nukleærmedisin som er ansvarlig for pasientbehandling i klinikken ved PET-senteret har ansvar for strålvern for ansatte som jobber med pasientbehandling.

Skjerming og sikkerhetssystemer

Det er ved syklotronen og ved produksjon av legemiddel i hotceller, fare for akutt bestråling om personale får tilgang til områder med høye aktivitetskonsentrasjoner. Sikkerhetssystemer er installert for å hindre at operatører kan utsettes for høye doserater. Gode skjermingstiltak for den daglige produksjonen skal bidra til lave doser for personalet.

Syklotronen har vegger og tak med 1,7 meter tykk betong for å skjerme omgivelsene mot ioniserende stråling, vegg mot terreng har en tykkelse på 40 cm. Taket til syklotronhvelvet er laget av 5 demonterbare betongplater som overlapper hverandre for å unngå lekkasjestråling. Betongen er av kvalitet B20. Alle kabel- og ventilasjons gjennomføringer til syklotronhvelvet er utført med skråvinkel for å hindre lekkasjestråling fra syklotronhvelvet. Hele produksjonsområdet som er radiologisk kontrollert område er skjermet med 20 cm betongvegg mot andre arealer ved PET-senteret. Legemiddelproduksjon foregår i hotceller levert av Von Gahlen, med 7,5 cm blyskjerming i alle retninger.

Sikkerhetssystemer som er etablert i produksjonsområdet ved PET-senteret:

- Radiologisk interlock tilkoblet alle hotceller i produksjon og R&D. Dette hindrer at hotcellene kan åpnes dersom det er radioaktivitet over et valgt nivå i hotcellene. Funksjonen kan overstyres med nøkkel dersom det er behov for det.
- PLS Target Switch system installert for sikker overføring av produkt fra syklotron til hotcelle. Systemet sørger for at radioaktivitet ikke kan overføres fra syklotron til en åpen hotcelle.
- Syklotrondør tilkoblet interlock system ved syklotronen. Syklotronen kan ikke startes med åpen dør, og åpnes døren under produksjon vil syklotronen stoppe automatisk.
- Radiologisk overvåking av laber og utslipp installert. Syklotron hvelv, Syklotron Access, Cleanroom og R&D-lab er kontinuerlig overvåket av gammadetektorer i rommene. Alarm på disse vil gi lokal lyd og lys alarm. Avtrekksluften fra hotceller og syklotron hvelv er overvåket av plastdetektorer. Alle måleverdier logges og presenteres på pc i kontrollrom. Systemet er av typen Thermo FHT 6020 og er levert, installert og testet av Laborel.
- Adgangskontroll med kode til produksjonsområdet.
- Måleutstyr for måling av radioaktivitet. Måleutstyret er valgt ut fra energi og måleområdet som er aktuelt ved PET-senteret.
- Waste gas system installert i syklotron hvelvet. Systemet forsinker avfall av produsert radioaktiv gass tilstrekkelig til at aktiviteten er dødd ut når gassen slippes ut.
- Lokal blyskjerming i QC-lab og Syklotron hvelv.
- Tilgjengelig verneutstyr.
- Wolframbeholdere for transport av FDG.
- Varselsystem med skilting og lamper for ”beam on” ved syklotron.

- Nødstopp.

NMS som er eier av bygg og utstyr har tilrettelagt for sikker håndtering av radioaktivitet ved å installere skjerming og sikkerhetsutstyr i klinikkdelen ved PET-senteret.

Skjerming og sikkerhetsutstyr etablert i klinikken ved PET-senteret:

- 20 cm betongskjerming i vegger rundt injeksjonsrommene og hotlab, samt 2 mm bly / 2 mm blyekvivalens i dører til injeksjonsrommene.
- 8 mm blyskjerming i tillegg til 20 cm betong i vegg mellom injeksjonsrom og venterom.
- 10 mm blyekvivalens i gjennomstikkskap fra injeksjonsrom til hotlab.
- PET/CT skanner skjermet med 20 cm betong i vegger mot oppholdsrom og 15 cm betong rundt resten, i tillegg til 2 mm bly i alle vegger. Dørene er skjermet med 2 mm bly og vindu inn til kontrollrom har 2 x 2 mm blyekvivalens.
- Blyskjermet sikkerhetsbenk i Hotlab, for opptrekking av pasientdoser.
- 2 mm blyekvivalens i vindu rundt ekspedisjonen.
- Egen transport tralle for sikker transport av wolframpotter med aktivitet fra produksjonen.
- Måleutstyr tilgjengelig.
- Radiologisk overvåking av Hotlab og PET/CT skanner på lik linje som ved produksjon. Dette systemet er det samme som for produksjon og er koblet sammen.

Monitorering av strålingsskjerming

Ved oppstart av produksjon ble strålenivåer målt, i første omgang rundt syklotronhvelvet. Deretter ble hotcellene testet ved først å overføre en liten mengde aktivitet for så å øke denne gradvis samtidig med måling av doserate rundt hotcellene. Strålingslekkasje er definert som målepunkter hvor doseraten gjennom skjermingen er større enn dobbelt av referansedoseraten.

Ved oppstart på syklotronen ble det funnet noen svakheter i stråleskjermingen rundt syklotronen. Dette utenfor ventilasjonsgjennomføring til syklotronhvelvet, hvor det ble målt forhøyet stråling fra gamma og nøytron. I Cleanroom kunne en se et noe høyere strålenivå innerst i rommet på siden av hotcellene, området ble sperret av og merket. Dette er det området som er nærmest target området, hvor strålingen er mest intensiv i syklotronhvelvet. Strålingslekkasjen gjennom ventilasjonskanalene derimot krevde et omfattende arbeid med tilleggskjerming i syklotron hvelv.

Det ble også avdekket en rekke punkter med strålingslekkasje rundt hotcellene. Dette ble utbedret av hotcelle leverandør Von Gahlen. Lekkasjene var rundt blyglass, ved dørhengsler og ved dørtetting, og strålingslekkasjen var i størrelsesorden 3-180 µSv/t. Lekkasjen rundt blyglass ble tettet med blyull og tilleggskjerming av bly ble montert på hotcellene ved lekkasjepunkter på hotcelle dører.

Resultat av doserate måling ved PET-senteret etter tilleggskjerming av hotceller og syklotronhvelv, er gitt i tabell 1.

Tabell 1 *Målte strålenivåer ved PET-senteret*

Målested	Arbeidsoperasjon	Doserate gamma [µSv/t]	Doserate nøytron [µSv/t]
Syklotron acess, inntil tilluftskanal	^{18}F produksjon ved syklotron, 30 µA/30 µA dual target	35	18
Syklotron access ved barriere		10	7
Syklotron access ved dør		3	0
Syklotron access		1,5	-
Clean Change i vinkel fra avtrekkskanal		10	3
Clean change ved dør		5	-
Clean change ved barriere		0,5	-
Clean room til venstre for hotcelle, inntil vegg mot syklotronhvelv		0,7	-
Clean room til høyre for hotcelle, inntil vegg mot syklotronhvelv		7	-
Clean room foran hotceller, ved betjeningspanel		2	-
Lockers		1	-
Control office		0,3	-
Korridor		0,3	-
På taket av syklotronhvelvet		2,6	-
Garderober inntil syklotronhvelv		0,3	-
Cleanroom foran syntese hotcelle	Syntetisering av 50GBq ^{18}F i hotcelle	0,7	-
Cleanroom foran dispenserings hotcelle	Dispensering	0,5	-
Cleanroom	Doserate ved uttak av produkt	5	-
Sluse til R&D-lab	Analyse av produkt i QC-lab	1,8	-
R&D-lab foran hotceller	150 GBq ^{18}F i hotcellen	0,5	-
Hot lab klinikk	Ved opptrekking av dose i blyskjermet sikkerhetsbenk	10	-
	Wolframpotte med 6 GBq plassert i skjermet posisjon i sikkerhetsbenk	0,4	-
	Injeksjon av pasient i injeksjonsrom ved siden av hot.lab	0,8	-
Kontrollrom PET/CT skanner	CT av injisert pasient i skanner	0,4	-
Venterom	Injisert pasient i injeksjonsrom	0,3	

Målinger er utført med Canberra Radiagem og NE Neutron Meter måleverdier er brutto doserate. Bakgrunn stråling for gamma er 0,3 µSv/t.

Tilleggskjerming av syklotronhvelv

Ved produksjon av 6 Ci (220 GBq) ^{18}F, som er regnet for maks kapasitet hva syklotronen kan levere og den produksjonen ved syklotronen som avgir mest nøytronstråling, ble det avdekket lekkasje av nøytron- og gammastråling gjennom to ventilasjonskanaler fra syklotronhvelvet og ut til Syklotron Access. Målingene viser en tydelig beam i samme vinkel som ventilasjonskanal. Ventilasjonskanalene er laget på skrå etter anbefaling fra GE Technologies for å unngå strålingslekkasje. Uheldig plassering av ventilasjonskanaler i forhold til plassering av syklotron i hvelvet er en mest sannsynlig årsak til lekkasjen. Lekkasjen berører Syklotron Access som er et vedlikeholdsområde utenfor syklotronhvelvet.

Det ble etter anbefaling fra Ge Technologies montert 20 cm tykke PEHD 500 (høymolekylær polyetylen high density) plater i underkant av ventilasjonskanalene. Doseratemålinger på lekkasje punktet ble målt på nytt etter skjermingen men viste liten effekt. For nøytroner fikk vi redusert strålingen med 25 %, for gamma så vi nesten ingen effekt. Det ble besluttet å skjerme ytterligere med PEHD 500 og bly i syklotron hvelv. Det ble denne gangen lagt 5 cm blyblokker oppå polyetylen platene og bygget skjerming rundt ventilasjonskanalene med PEHD 500. Målingene viser at skjermingstiltakene som er utført har redusert lekkasjen gjennom tilluftkanalen med 91 % for nøytroner og 56 % for gamma. Gjennom avtrekkskanal er lekkasjen redusert med 92 % for nøytroner og 75 % for gamma. Måleresultat før og etter skjerming er gitt i tabell 2.

Tabell 2 *Doseratemålinger før og etter tilleggskjerming i syklotronhvelvet*

Målested	**Før skjerming Mai 2006**		**Etter skjerming Februar 2007**		**Etter skjerming August 2007**	
	Doserate gamma [µSv/t]	**Doserate nøytron [µSv/t]**	**Doserate gamma [µSv/t]**	**Doserate nøytron [µSv/t]**	**Doserate gamma [µSv/t]**	**Doserate nøytron [µSv/t]**
Syklotron Access, inntil tilluftskanal	80	200	77	150	35	18
Syklotron Access ved barriere (tilluftskanal)	30	90	22	30	10	7
Syklotron Access ved dør (tilluftskanal)	7	0	-	-	3	0
Clean Change i vinkel fra avtrekkskanal	40	40	22	20	10	3
Clean Change ved dør (avtrekkskanal)	6	5	-	-	5	0

For å vurdere om skjermingstiltakene var tilstrekkelig eller om ytterligere tiltak måtte iverksettes ble det utført en vurdering av dose til operatør i ulike scenarioer. Beregninger har vist at operatører ved produksjon mest sannsynlig vil få en dose < 20 µSv per år som følge av denne lekkasjestrålingen. Det ble derfor besluttet at det ikke var nødvendig med ytterligere skjerming, men tiltak som merking og opplæring ble iverksatt og skal sikre lave doser.

Konklusjon

Ved planlegging og bygging av nytt PET-senter har det vært mulig å tilrettelegge for lave doser for personalet og for allmenn befolkning. Skjerming og sikkerhetssystemer tilpasset aktiviteter ved PET-senteret er installert. Kontroll av skjerming har avdekket strålingslekkasje fra hotceller og syklotronhvelv. Strålingslekkasjer er tilleggsskjermet og måling viser lave strålenivåer ved kontrollerte områder ved PET-senteret. Sikkerhetssystemer som skal sikre operatører mot å utsettes for høye aktiviteter er testet. Strålingslekkasjer vi har kunnet avdekke ved vårt produksjonsanlegg understreker viktigheten av å gjøre en grundig kontroll av skjerming og testing av sikkerhetssystemer. Lave doserater i produksjonsområdet og gode arbeidsprosedyrer gjenspeiler lave persondoser for operatører.

Referanse

1. GE Medical Systems PETtrace Site Planning Guide, Direction 2102960-100, rev 4
2. Basisprosjekt Rikshospitalet-Radiumhospitalet HF, PET-senter. Norconsult 15.12.2005

Vedlegg 1: Oversiktstegning av PET-senteret med radiologisk klassifisering

Software and algorithms for online medical dosimetry with luminescence detectors

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Abstract

Luminescence detectors such as organic scintillators or crystals of carbon-doped aluminum oxide (Al_2O_3:C) can be used for online in vivo medical dosimetry. The main signals from aluminum oxide are radioluminescence (RL) and optically stimulated luminescence (OSL). The detectors can be attached to optical fiber cables and they can therefore be read out remotely, for example, during radiation therapy of cancer patients. The prime application is so-called dose verification during treatments such as brachytherapy or external-beam radiotherapy with high-energy x-rays. The objective of dose verification is to provide an independent measurement of the planned dose. Essentially dose verification can therefore help prevent radiation accidents. The aim of this presentation is to give some highlights from this work with an emphasis on the developed software and algorithms. The prime software is programmed in LabView from National Instruments and it can handle data acquisition, online dosimetry calculations, interactive calibration, uncertainty analysis and presentation of key results during the treatment. Additional data analysis is carried out using a dedicated library of functions programmed in S-plus from Insightful Inc.

Introduction

Risø has been engaged in the medical applications of fiber-coupled luminescence dosimetry since 2001 and we have collaborated closely with, for example, the university hospitals in Malmø, Århus, and Copenhagen. The main work has concerned proton dosimetry, dosimetry for brachytherapy, IMRT dosimetry, and dosimetry for diagnostic radiology.

Many measurements have been carried out at Risø using radioactive sources or small x-ray generators as a substitute for clinical beams. However, numerous field trips have been made to “real" clinical hospital environments, and measurements have often been carried out during evenings, nights or weekends when there were no patients around. Upon arrival, we roll out two or three 15 m optical-fiber dosimeter probes and we set up our luminescence detection equipment. Beam alignment is always needed and we need to understand how the beam is controlled and why certain interlocks occur etc. Some confusion always seems to prevail (at least) during the initial first hour of any of such field trip. If the available beam time is very limited, it is therefore particularly important that the acquisition software is both reliable and flexible. On the one hand, we need to be able to modify the measurement protocol to adapt to any unforeseen local condition, and on the other hand, it is always important that the software is reliable such that we get the data in a controlled and well-defined way according to a standard protocol. To get a feeling for the quality of the measurements, we furthermore need some capability for online calibration and analysis.

An important part of the solution to these problems has been to develop special software in the Labview graphical programming language. The main software consists of two stand-alone Microsoft Windows applications - one for acquisition and another for analysis. The following outlines the dosimetry system in more detail and gives three specific examples where the tailored software has been important.

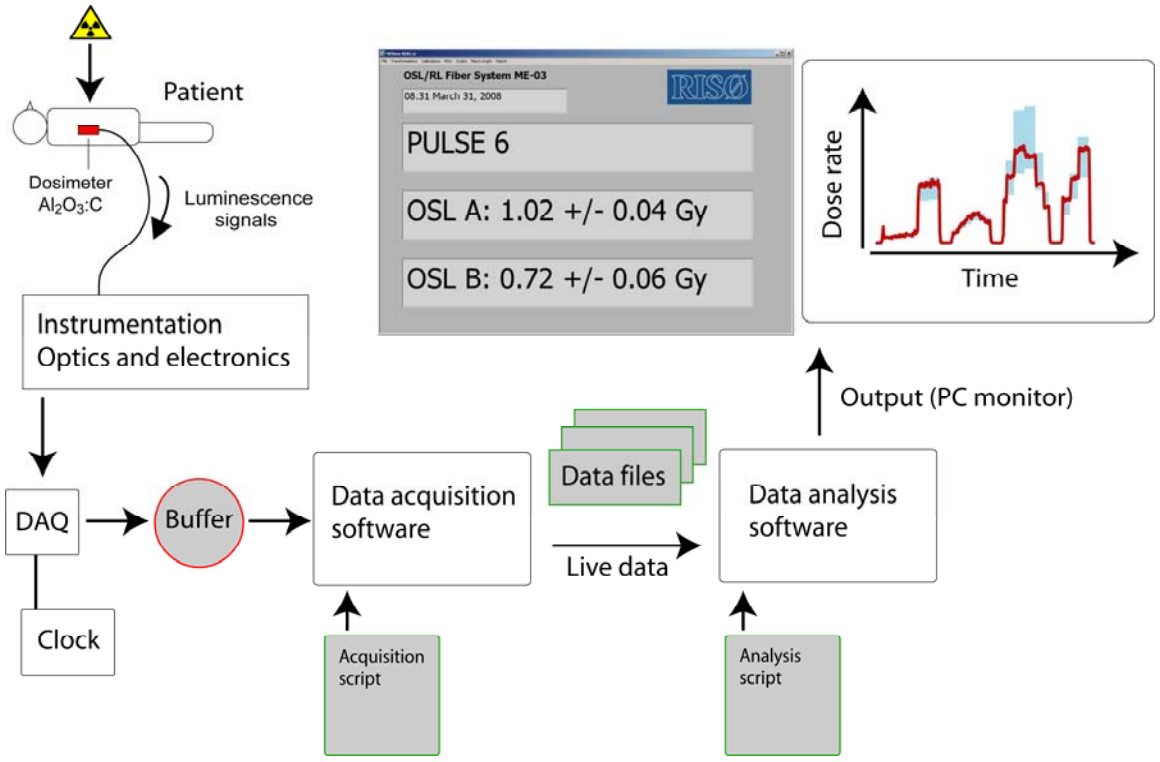

Figure 1 *Online in vivo luminescence dosimetry system.*

System

Figure 1 shows the dosimetry system. The luminescence signals (either radioluminescence or optically stimulated luminescence from aluminum oxide or fluorescence from organic scintillators) [1] are guided to the instrumentation by optical fiber cables (1 mm in outer diameter). The luminescence is monitored with low-background photomultiplier tubes run in single-photon counting mode, and the data acquisition system records the light intensity using so-called event counting. Other analog and digital signals related to, for example, laser control, temperatures, and time-resolved counting are also logged. Due to the nature of the clinical applications, we have designed the system with the aim to achieve a reproducibility better than 1 % (one standard deviation).

Timing is critical for the data acquisition and control. It is vital that if we set the system to record the luminescence with, for example, 1000 readings per second, then we truly get one single reading every 1 ms without any significant jitter from reading to reading regardless of what other tasks are taking place on the computer running the data acquisition software (e.g. virus scanning or hard drive tasks). The solution is to let the timing be controlled by a separate hardware clock and to let the data acquisition card (DAQ in Figure 1) output its results directly to a buffer in memory independently of

the CPU workload. This solution provides perfect timing in as far as the buffer is emptied occasionally.

Another important feature of the developed data acquisition software is that it can be controlled from a job file. Essentially the software has a scripting language such that the entire protocol for any measurement task can be specified in an ASCII file. Such scripts are in a human readable format and they therefore also serve as documentation for the measurement protocol.

Online in vivo brachytherapy dosimetry

In a project with Aarhus University Hospital [2], we performed online in vivo dosimetry in cervical cancer patients undergoing pulsed dose rate brachytherapy with a Varian GammaMed Plus Ir-192 afterloader. The treatments had durations from 10 to 50 hours with one single ~15 minutes pulse delivered every hour. An important task of the data acquisition software in this case was to automatically start the optical stimulation after each pulse had been completed. This task could relatively easily be accomplished by writing a job file that could simply synchronize the measurement cycle with the irradiation pulse as identified online from the recorded luminescence signals. This solution provided monitoring of the dose-rate using the Al_2O_3:C radioluminescence signal without interference from the laser light needed for the subsequent OSL-readout of the total dose. The procedure accounted for patient interlocks and other treatment delays.

Randomized laboratory tests

To quantify the influence of temperature variations on clinical Al_2O_3:C RL and OSL measurement results, we conducted an automated laboratory experiment over a 12 day period involving three-fold randomization of (1) irradiation temperature (10-45ºC), (2) stimulation temperature (10-45ºC), and (3) irradiation dose (0 to 4 Gy; 50 kV x-rays). We derived linear RL and OSL temperature coefficients using a simple statistical model fitted to all data (N=909). The study [3] showed that the temperature coefficients were independent of dose and other variables studied. We found that the RL signal changed only with irradiation temperature whereas the OSL response changed with both irradiation temperature, stimulation temperature and OSL integration time. Typically the temperature coefficients were of the order of 0.2 %/K, and these thermal effects are therefore large enough to be of importance for clinical measurements.

Figure 2 shows as an example the raw data (left panel) and the data after correction for temperature variations (right panel). The experimental standard deviation of the OSL response to 2.7 Gy dose is reduced from 1.3 % for the raw counts to 0.3% for the temperature corrected data (as indicated in the figure).

The experimental design (using threefold randomization and hundreds of readouts over a relatively long period of time) means that the influence of variations (both random and systematic) of the laser intensity, x-ray dose-rate and other important elements on the estimated temperature coefficients will tend to cancel out.

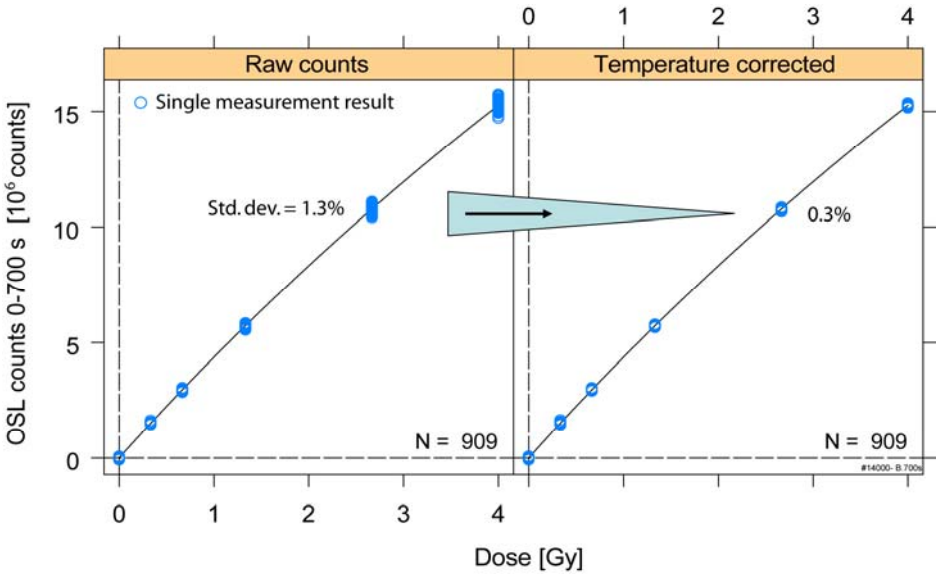

Figure 2. *Dose-response curve for randomized OSL readouts before and after temperature correction. Each panel includes 909 separate data point (circles), but due to the low scatter (especially for the temperature-corrected data), these points essentially fall on top of each other within the six dose levels.*

The job file for this experiment consisted of a single loop that was repeated 909 times. For each loop cycle, the software would draw three random numbers to determine what dose should be given, and what temperature should be used during the irradiation and during the OSL readout, respectively. It was important to randomize the testing to get a realistic view on the reproducibility since the crystal has a memory.

RL calibration

The RL-signal from Al_2O_3:C is relatively complex. The main problem is that the RL-sensitivity is not constant (as for an organic scintillator) where the RL-sensitivity is here defined as the RL count rate per dose rate unit. However, these sensitivity changes seem to follow a highly reproducible pattern, and a simple algorithm can used to obtain a precise real-time dose rate estimate from the RL signal [4].

Figure 3 shows as an example RL results from 175 MeV proton irradiation of an aluminium oxide crystal at the The Svendberg Laboratorium in Uppsala, Sweden [5]. The crystal was given 8 Gy in 20 shots of 0.4 Gy each. The crystal was placed at 60 mm depth in a water phantom (i.e. in the plateau of the Bragg curve). Part A of Figure 3 shows the raw RL results. The increase in sensitivity is clearly seen in the graph: the count rate increases from pulse to pulse. To calibrate the system, we need to calculate a dose-response curve as shown in part B of Figure 3 [6]. This effectively means that we have to identify and integrate each of the 20 pulses seen in part A. This can be a relatively tedious work, but with the new data analysis software we can either perform this calibration interactively or we can run a pre-programmed script that automatically identifies the 20 pulses and creates a calibration table. The same script can subsequently apply the RL-algorithm and convert any new data to dose rates.

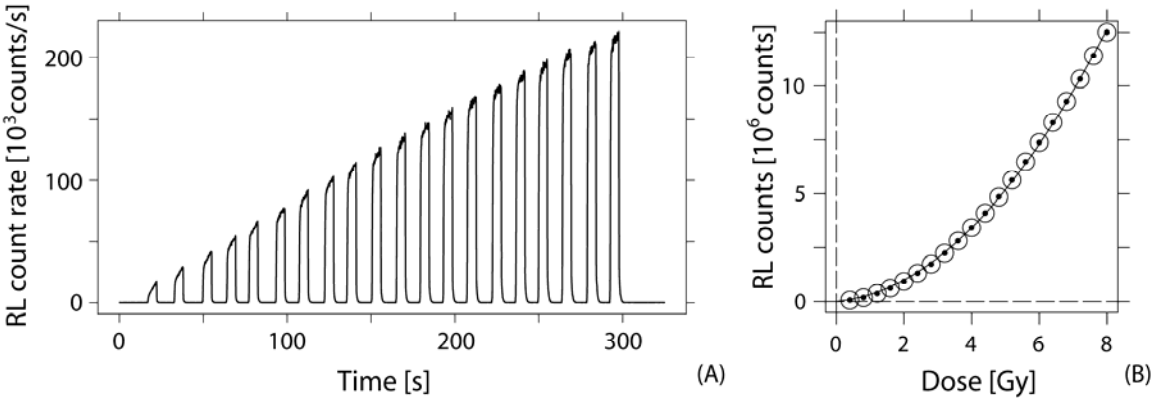

Figure 3. *(A) Raw RL count rate versus time for 20 x 0.4 Gy proton irradiations as described in the text. (B) shows the corresponding RL dose-response curve. We note, for example, that 7 Gy give a total of about 10^7 counts.*

Acknowledgment

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1 POSTER

Innovative Technology and Instrument to Explore the Risk of Applying High Gamma Radiation Doses on the Nanostructure Surface of the Bones

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Abstract

Gamma radiation is used for radiation therapy to treat carcinogenic diseases, including bone cancer. However, there are side effects by radiation on structures of bones. Structure alters electron density of states that, in time, influences biomedical reactions on bone life condition. Exoelectron emission (EEE) phenomenon underlies an electron spectroscopy to explore alteration of the electronic structurally influenced properties of materials. The l method gives am opportunity to explore imperfections at surface layer. . There are just a few reports on EEE application to study bones. Because of the above, this study is devoted to EEE application for exploring gamma radiation induced structural explorations in bones. The main objective of this study is to develop EEE measurement technique for investigating gamma irradiated bones. Influence of γ-radiation applied for radiation therapy on bone EEE has been recognised.

Introduction

Bone is the main component of the skeleton in the adult human. Like cartilage, bone is a specialised form of dense connective tissue. Bone gives the skeleton the necessary rigidity to function as attachment and lever for muscles and supports the body against gravity [1]. Bone is composed of collagen, water, hydroxyapatite mineral and small amounts of proteoglycans and noncollagenous proteins [1]. Some data in the literature for the composition of adult human and bovine cortical bone are given in Table 1.

Table 1. ***Comparison in*** *composition of Adult Human and Bovine Cortical Bones [2].*

Component	Bovine	Human
Water	9.1%	7.3%
Apatite mineral	67.4 %	67.4%
Organic matrix	21.5%	21.4%

The electron structure of bone and its surface vary under different treatment. Several electron canters with maximums at -5.0eV, -5.3eV, -5.7eV were found in bone tissue [1].
Because gamma radiation is widely used in therapy as well as in bone cancer therapy, the effects of gamma radiation on human bones should be investigated. The bovine cortical bone specimens can be used as prototype of adult human bone because of similarities in compositions of adult human and bovine cortical bones.

Methodology

Exoelectronemission (EEE) was developed as an electron emission for different man-made materials analysing. This technique is extremely sensitive to structural imperfections in material surface [3]. EEE is a radiation of low-energy electrons from the surface of a firm body being in the non-equilibrium excited by any way condition [4]. The common advantage of EEE is exploring the influence of structural changes in the surface and subsurface region (depth of about 10 nm or less on the emission of electron [4].

The device used to detect EEE consists of a vacuum system, a complex of electronic devices to detect electrons and process the signal and blocks of a photo- and thermo-stimulation of the specimen.

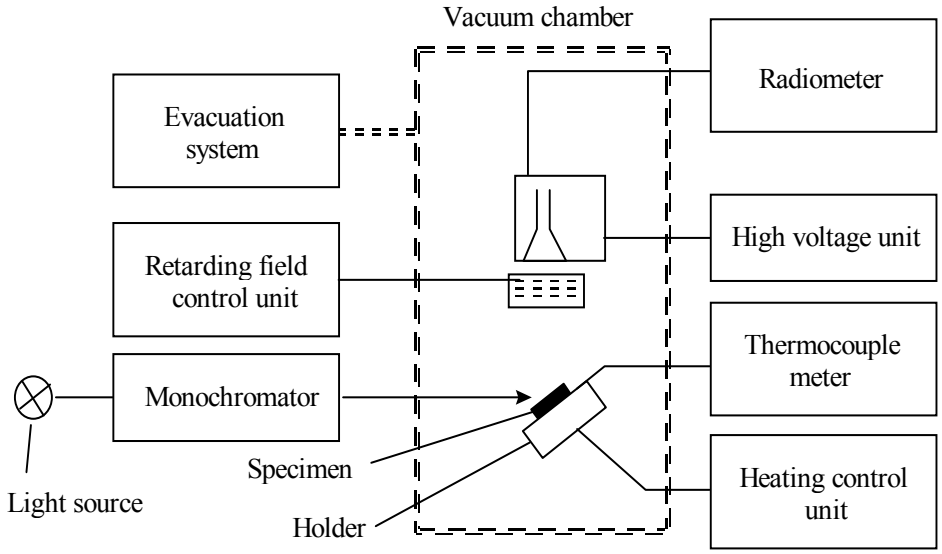

Figure 1. *The scheme of the Exoelectron Spectrometer*

The specimens were prepared from a bovine tibia bone of slithered 1.5 years old cow. The bovine bone was bought after 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 [3]. Specimens were cut in a way so that they 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.

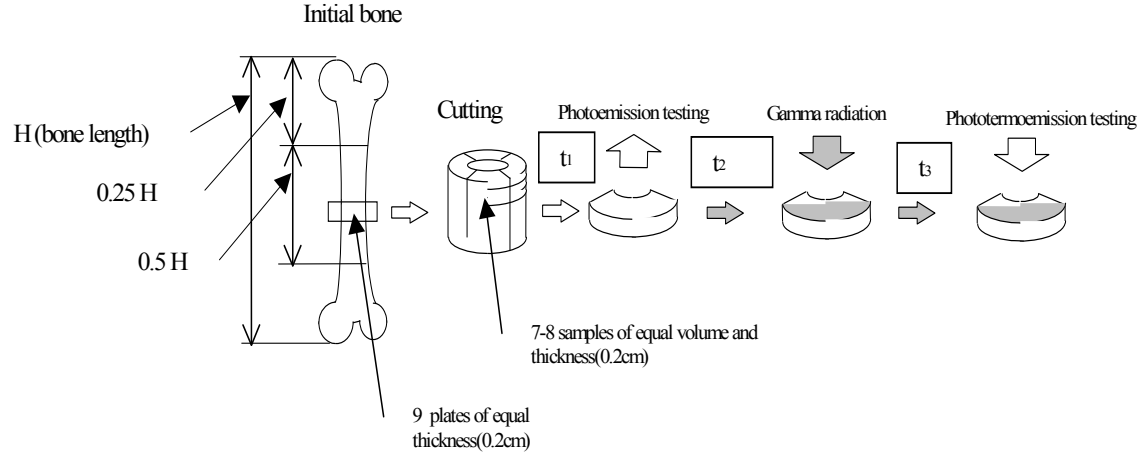

Figure 2. *Technology of preparation and emission testing of the model specimens*

Some factors that might influence our final results were considered. These factors are: duration of bone storage, temperature dependence, heating rate conditions, and photon energy.
After several experiments had been applied, we realised that the technology and instrument on EEE measurements have been reached to get the highest exoemission maxima and reproducible results when:
- Exoemission measurements should be conducted after the 10th day of bone storage;
- The energy of photons should be equal to 5.6 eV;
- Measurements should be done at the 1st heating cycle;
- Rate of heating should be equal to 10 °C/min;

The uncertainty of Photo-thermo-stimulation emission (PTSE) registration that is conditioned by instability of the whole systems of the device were evaluated as well as statistics of electron emission and statistics of electron emission. Uncertainty is calculated for non-direct measurement case. The uncertainty associated with statistics of emission current is equal to $\sqrt{I}$, where *"I"* is the emission current. However, the value of the *"I"* depends on the physical processes within the material and its instability. For verification of the uncertainties, there are two steps required:

First step is to calculate uncertainty of δI of I (EEE current) and then second one is to verify the regularity of relative uncertainty (δQ) on Dose (D), Energy (E), and dose rate.

$$\delta F = \frac{\sum 0.5(\Delta I_i + \Delta I_{i+1})(T_{i+1} - T_i) + I(T)\left[\Delta T_{i+1} + \Delta T_i\right]}{\sum I(T)(T_{i+1} - T_i)}$$

Where:
I_0 is the initial current at room temperature before heating;
T_i is the temperature for $I(T_i)=I_i$;
T_{i+1} is the temperature for $I(T_{i+1})= I_{i+1}$;
$T_{i+1} - T_i \geq \Delta T$, where ΔT is temperature error

Because ΔT is small and $I_i \rightarrow I_{i+1}$, we can assume that

$$\delta F = \frac{\sum 0.5(\Delta I_i + \Delta I_{i+1})(T_{i+1} - T_i) + I(T)\left[\Delta T_{i+1} + \Delta T_i\right]}{\sum I(T)(T_{i+1} - T_i)}$$

Where $\Delta T=const\approx 0.3^0C$
To estimate (δQ) as the $\Delta Q/Q$, the below formula was used:

$$\delta Q = 1/\sqrt{I_0} + \frac{\sum \left[\sqrt{I_i^*}(T_{i+1} - T_i) + I_i^* \cdot \Delta(T_{i+1}) + \Delta T_i)\right]}{\sum I_i^*(T_{i+1} - T_i)}$$

Where $\Delta T=const\approx 0.3^0C$

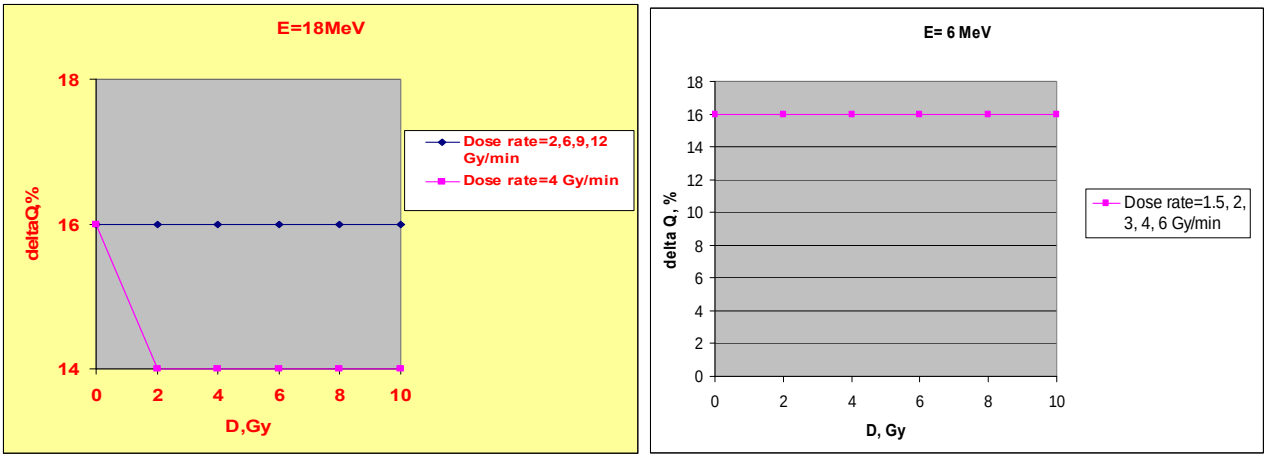

Figure 3 *Uncertainty ΔQ for Energy (E) of 18 and 6 MeV*

Using the Student Coefficient t-test (significance 95%), we have proved that the ΔQ differences for different irradiation modes are not significant ΔQ≤ 16%.
Gamma radiation was delivered from the linear accelerator SL75-10 and Variance Clinac 2100 C/D Philips SL75-10 medical type linear accelerator provides gamma photons with fixed 8 MeV energy, while Variance Clinac 2100 C/D for 6 and 18 MeV X-ray photons. The beam is pulsed with duration of 2 microseconds and repetition rates selectable between 100 and 600 per second. In the photon mode the dose rate is approximately 10 Gy (1 kilorad) per minute measured at 1 metre from the target. In the electron mode, dose rates in the region of 10 kGy/ minute at 1 metre are possible. This is equivalent to ~ 0.3 Gy (30 rads) per pulse.

Table 2. *Plan for Gamma irradiation of bone specimens*

Irradiation mode	Number of specimens	E, Mev	D, Gy	D, Gy/min
Fractional				
Experiment #1	27	8 (typical for radiotherapy)	2,4,6,8,10	1,2 (typical for radiotherapy)
Non-Fractional				
Experiment#2	69	6	2,4,6,8,10	2,3,4,6
Experiment #3	27	8	2,4,6,8,10	2
Experiment #4	27	8	1,2,3,4,5,6,7,8,9,10	2
Experiment #5	69	18	2,4,6,8,10	1,2,4,6

Results and discussion

Figure 4 shows the Q behaviour for a bone material after radiation by gamma photons with Energy E= 6MeV. Radiation had been performed with different dose rate (2 Gy/min and 6 Gy/min respectively).

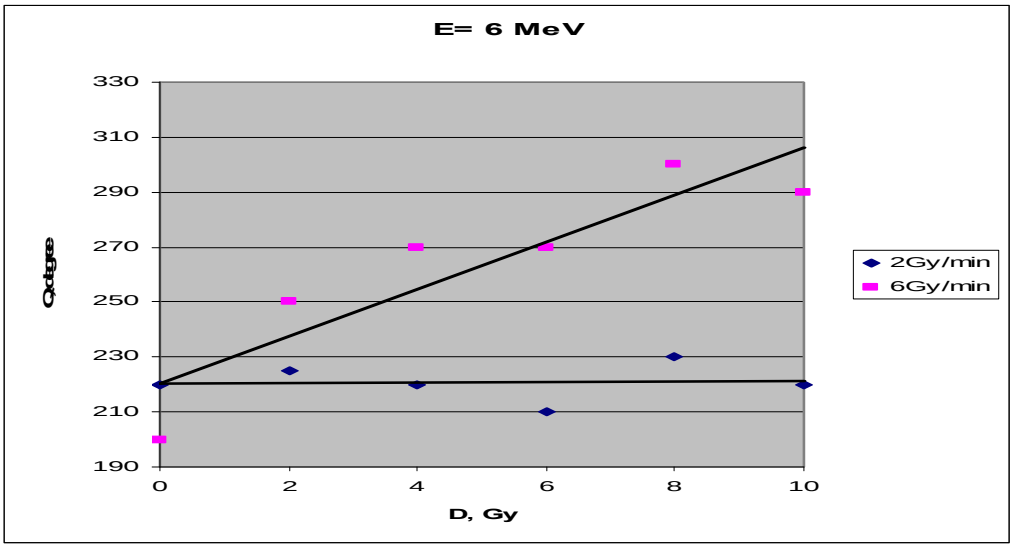

Figure 4. *Dependence of the Q average values (3-5 measurements for each dot) on gamma radiation absorbed dose (D) of gamma photon for E = 6 MeV and absorbed dose rates ($\dot{D}$ min= 2 Gy/min and $\dot{D}$ max= 6Gy/min)*

Following the data in Fig.4, one can conclude that EE has a response when the dose rate is 6 Gy/min but does not have the response at 2 Gy/min.

Figure 5 shows the Q behavior of bone material after irradiation by gamma photons with energy E= 18 MeV. Radiation had been performed with different dose rate (2 Gy/min and 6 Gy/min respectively).

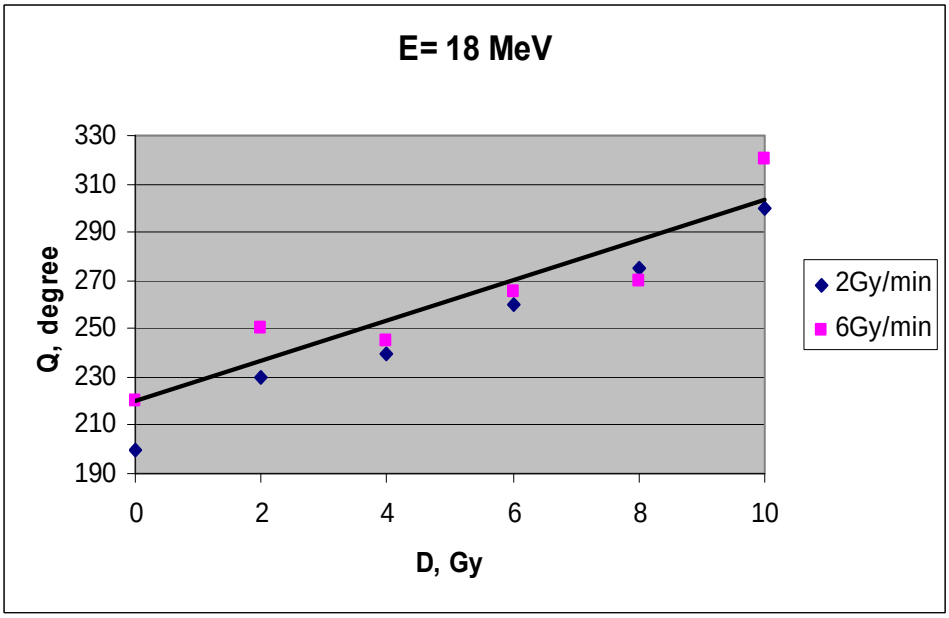

Figure 5 *Dependence of the Q average values (3-5 measurements for each dot) on gamma radiation absorbed dose (D) of gamma photon for E = 18 MeV and absorbed dose rates ($\dot{D}$ min= 2Gy/min and $\dot{D}$ max= 6Gy/min)*

The data on Fig.5 give the evidence that EE responses for the dose rates 2 Gy/min and 6 Gy/min are similar and demonstrates linear correlation.

General conclusion

1- Technology of specimen preparation has been developed;

2- Technology and instrument of EEE measurements have been done to get the highest exoemission maxima and reproducible results:

* Exoemission measurements should be conducted after the 10th day of specimen preparation;

* The energy of photons should be equal to 5.6 eV;

* Measurements should be done at the 1st heating cycle;

* Heating rate should be equal to 10 °C/min;

3- Uncertainty of Q (exoemission signal detection) does not exceed 16%;

4- Both fractional and non-fractional modes at 8 MeV have similar impacts to the exoemision signal Q;

5- Gamma Radiation increases exoemission signal Q (concentration of radiation induced imperfections) within the dose range of 2-10 Gy:

- For Non-Fractional Radiation modes:

18MeV, 2, 6 Gy/min,

8MeV, 2 Gy/min,

6 MeV, 6 Gy/min,

- For Fractional Radiation mode:

6 MeV, 1, 2 Gy/min

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3 Effect studies session

Nordic collaboration within biological dosimetry

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Abstract

Biological dosimetry is presently performed in the Nordic countries in two laboratories: at STUK-Radiation and Nuclear Safety Authority and in the last few years also at FOI, The Swedish Defence Research Agency. Both Norway and Denmark have an informal agreement with STUK to perform biological dosimetry on their behalf. However, in the event of mass casualties, the capacity of a single laboratory could be overwhelmed and the collaboration between countries is vital. From 2005, FOI, STUK and NRPA have been working together to optimize a method for biological assessment of radiation dose for specific application in emergency preparedness. A two-year NKS-funded BioDos project has aimed at improved methods for biodosimetry that have specific application in emergency preparedness. An assay in which chromosomes are prematurely condensed (PCC) was investigated. Chemically induced PCC provides a potentially faster means of analysis as well as the ability to assess higher doses than with the dicentric assay which is routinely applied in biodosimetry.

The experiments indicated that a protocol utilizing okadaic acid is adequate for induction of PCC cells in stimulated lymphocytes for the subsequent evaluation of ring chromosomes. The establishment of dose response curve and a preliminary comparison of the assay method to the traditional dicentric assay in triage mode have been performed. The triage method is based on limited analyses enabling a categorization to high, medium and low dose classes in an emergency situation involving large numbers of people.

The collaboration between STUK, FOI and NRPA continues in 2008 with the new NKS project BioPex that has the aim to test and evaluate the applicability of the PCC assay in a simulated triage exercise involving a large number of exposed casualties.

Introduction

A large number of radiation exposed subjects, including persons with reasonably high doses, can be expected in a range of scenarios including large scale accidents and those dealing with malevolent use of radiation. Following an event of this kind, there is a need for fast assessment of dose for a significant group of individuals from the public or emergency responders. Radiation protection authorities, health services, and defence agencies need access to competence to evaluate and reassure potentially exposed individuals. Rapid and reliable dose estimation would be necessary for medical decisions. The large number of casualties creates a challenge to the relatively small biodosimetry laboratories. Collaboration and networking are needed to provide fast and reliable dose assessment when the amount of samples exceeds the capacity of a single laboratory.

The Swedish Defence Research Agency (FOI), Radiation and Nuclear Safety Authority (STUK) in Finland, and Norwegian Radiation Protection Authority (NRPA) have worked together to optimize a method for biological assessment of radiation dose for specific application in emergency preparedness. This two-year NKS-funded BioDos project has aimed at improved methods for biodosimetry that have specific application in emergency preparedness. The prematurely condensed chromosome (PCC) assay was investigated for the purpose. The PCC assay provides a potentially faster means of analysis as well as the ability to assess higher doses than with the dicentric assay which is routinely applied in biodosimetry.

The technique was proposed for biological dosimetry purposes by Pantelias and Maillie (1984). The original procedure involved fusion of interphase lymphocytes with mitotic Chinese hamster ovary cells resulting in mitotic factors inducing the nucleus to condense into chromosomes. The analysis is based on the counting of excess chromosome fragments from Giemsa-stained preparations. This procedure includes a number of technical drawbacks, e.g. the decrease of sensitivity over time due to rejoining of radiation-induced breaks and the more demanding preparation and analysis of PCC cells than in the classical chromosome analysis. However, the PCC assay has important potential applications. In high-dose exposures, lymphopaenia reduces the number of cells available for chromosome analysis and the cells that are available have a low mitotic index due to severe cell cycle delay. In PCC assay, these difficulties are overcome by scoring of radiation-induced damage in resting cells. In recent developments of the assay, okadaic acid and calyculin A induced chromosome condensation of stimulated, proliferating cells enables scoring of condensed chromosomes (Durante et al. 1998; Kanda et al. 1999) up to a dose of 20 Gy. The combination of FISH chromosome painting and PCC assay has enabled recognition of chromosome exchanges (Durante et al. 1996). To avoid cell culturing, a further development of the assay for induction of PCC in unstimulated cells was facilitated by cyclin B kinase (Prasanna et al. 2000).

The paper describes the collaborative process of developing and testing the PCC assay for the purpose of biological dose assessment particularly for high doses and in the event of mass casualties. In its first phase, the project focused on comparing the available assays and the development of optimal conditions as well as analysis methods. In addition, development of scoring criteria for PCC rings, establishment of a PCC ring dose response curve, and a triage comparison of the PCC assay to the classical dicentric assay mode, were performed. A continuation of the project (BioPex) will aim at a large-scale testing of the PCC assay in parallel with the classical dicentric method in a simulated mass-casualty exercise.

Evaluation of PCC assay conditions and analysis methods

A variety of protocols for chemically induced PCC was tested. Phytohemagglutinin stimulated lymphocytes were cultured for 48 h in 37ºC and treated with either okadaic acid (OA) or calyculin A (Cal A), two phosphatase inhibitors inducing chromosome condensation in any cell cycle phase. In addition, experiments using cyclin B kinase together with either of the phosphatase inhibitors were performed for PCC induction in unstimulated lymphocytes, and would therefore offer a faster approach than the OA and Cal A assays. Routine harvest of the cultures was performed and the induction and quality of the PCC cells were estimated from Giemsa stained preparations. The experiments showed that a higher yield of PCCs were obtained using CalA, however, the quality of the PCCs was worse for CalA than for OA treated cells. Apparently for this reason, the dose response in aberration yields, especially for PCC rings, was more evident in cells treated with OA.

For the evaluation of different analysis methods, PCC preparations from cultures treated with OA and Cal A were stained with both Giemsa and fluorescence in situ hybridization (FISH). From the Giemsa stained preparations, scoring was focused on excess fragments and on ring chromosomes. Evaluation

of excess fragments was more laborious and included more uncertainty than the scoring of PCC rings. Although ring chromosomes were easy to distinguish, it turned out that clear definitions for the appearance of scorable rings were needed. Evaluation of rings that have clearly defined open spaces as well as clearly spherical rings without open spaces seems to provide the most reproducible results (Figure 1 and 2).

FISH analysis after whole chromosome painting of chromosomes 2, 4 and 12, with different fluorochrome combinations, was concentrated on exchange type aberrations. In these experiments, clearly distinguishable exchanges and fragments in the painted chromosomes were observed. The FISH approach appeared reliable for doses of 5 Gy and below, but higher doses imposed problems in classification of aberrations due to the high number of complex aberrations (Figure 3). For PCC cells from kinase treated cultures, only FISH painting was performed. The kinase method induced relatively low level of chromatin condensation and thus FISH painting resulted in painted chromosome "areas" or spots (Figure 4). Defining the spots in an accurate manner and distinguishing one spot from another emerged as the main difficulty for the kinase approach. While the method in principle affords a faster overall PCC induction, reliability in evaluations are limited and the method has some degree of technical difficulty.

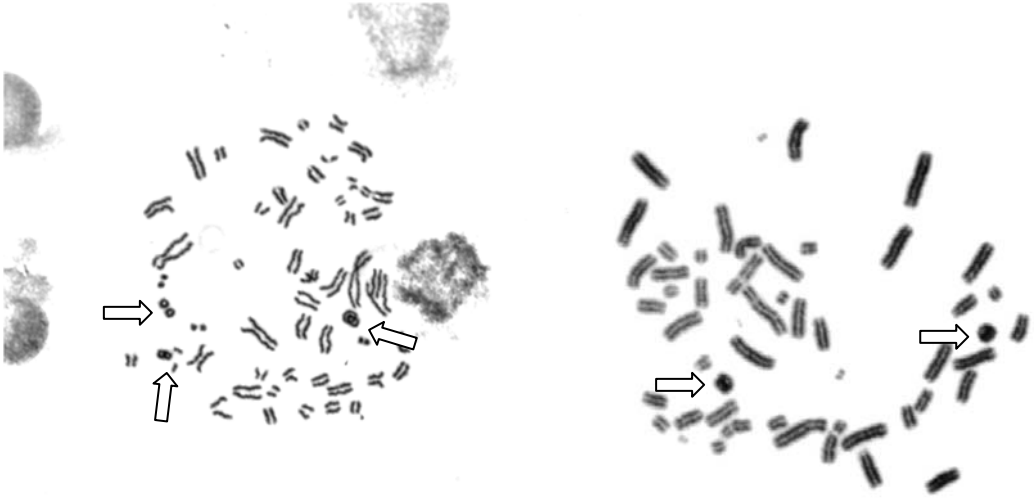

Figure 2. *OA induced PCC cell in M/A phase with two ring chromosomes (arrows).*

Figure 1. *Okadaic acid induced PCC cell in G2/M phase with three rings (arrows) after 7.5 Gy dose.*

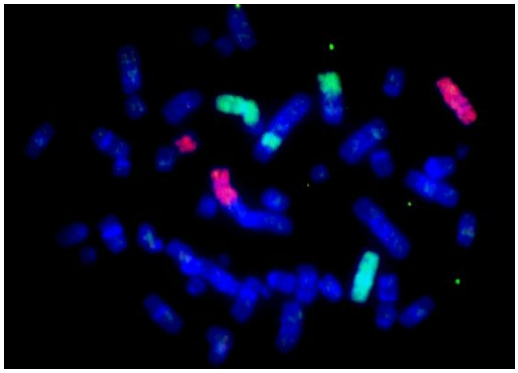

Figure 3. *FISH painted PCC-cell with multiple aberrations.*

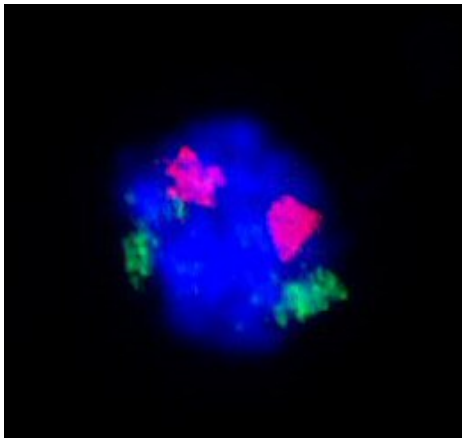

Figure 4. *Kinase cyclin B and Calyculin A induced chromatin condensation with FISH painting*

Dose response curve

Whole-blood was collected in vacuum tubes from one donor. Irradiations were performed in the Metrology Laboratory at STUK. The tubes were placed in 37 ºC water bath and irradiated with doses of 0, 2.5, 5, 10, 15 and 20 Gy using ^{60}Co gamma-rays at a dose-rate of 0.3 Gy/min. 48-h lymphocyte cultures were treated with OA at 47 hours and harvested. PCC-ring analysis was performed on blinded preparations by scorers from both STUK and FOI after analysis criteria were set and inter-comparison between laboratories was performed. The combined data set comprised of almost 10000 analysed PCC cells. Applying the method of iteratively reweighted least squares for curve fitting according to standard methods (IAEA 2001), the results were best fitted to a linear model, $Y=C + \alpha \times D$, where C = 0,002 (±0,0025) and α = 0,049 (±0,0062). Similar linearity of PCC rings has also been observed in a previous study (Lamadrid et al. 2007). In addition, a small scale dose estimation triage was performed among the partners for both PCC rings and dicentrics in order to compare the two methods. The triage analysis mode for dicentrics comprised of either 50 metaphases or 30 dicentrics, and for PCC rings either 300 cells or 50 rings per sample. The preliminary finding indicated that the PCC ring assay may be more applicable at doses above 5 Gy whereas the dicentric assay is accurate at lower doses.

Mass casualty exercise

The implementation of the assay into emergency preparedness purposes is now underway in the BioPex project. The aim is to test and evaluate the applicability of the assay in a simulated triage exercise involving a large number of exposed casualties in a scenario of malevolent use of radiation, e.g. a hidden γ-source in a public place. *In vitro* exposures of blood samples with a wide range of doses, including non-uniform exposures, will be performed, and parallel cultures for both PCC ring and dicentric assays will be established. Dose estimation will be performed using the PCC dose response curve established in the BioDos project as well as the routinely used dicentric curves in the respective laboratories.

The overall goal of the simulated mass casualty exercise is to evaluate the applicability of the PCC ring assay in comparison to the dicentric assay. This is achieved by comparing the PCC and dicentric assay on several levels: speed and easiness of scoring, accuracy of dose estimate and correct category assessment. Finally, the applicability of the assays for emergency preparedness situations will be evaluated.

Conclusions

In order to assess the best technique for dose estimation based on PCC induction, a variety of culture parameters were investigated during this work. The quality and quantity of PCC cells produced and the reliability during evaluation of different aberration frequencies were also assessed. With these aspects in mind, the most promising PCC assay for the purpose of fast dose estimation of a large number of casualties encompassing a wide range of doses is the method using okadaic acid (without colcemid) during a 48 hour cell culture and subsequent evaluation of ring chromosomes. The evaluation of rings that have clearly defined open spaces as well as large spherical rings without open spaces seems to provide the most reproducible results.

FISH chromosome painting applications do not seem well suited for an emergency preparedness method where fast analysis is critical. The complex nature of aberrations at higher doses induces difficulties in evaluation. Furthermore, although in principle a faster method, the kinase method for obtaining PCCs did not produce high numbers or high quality PCCs, and evaluations with FISH techniques did not result in reliable evaluations to any degree.

The PCC dose response data seems to indicate that a linear fit best describes the data and that the PCC assay may be most applicable and relevant at doses 5 Gy and above. The dicentric assay is known to be reliable only up to 5 Gy, since saturation of aberration yield occurs at higher doses. On the other hand we have shown here that PCC rings follow a linear increment up to very high doses. Thus, for emergency preparedness applications, the dicentric assay and PCC assay cultures could be run in parallel, evaluated in triage mode by either assay to indicate the degree of the doses, followed by more detailed evaluation by the appropriate assay as time allows.

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Analysis of chromosomal aberrations in cells exposed to ionizing radiation for the purpose of retrospective dose estimation and the assessment of DNA damage

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Introduction

Although human cells possess efficient DNA repair mechanisms, some of the damage induced by ionizing radiation is misrepaired and leads to the formation of chromosomal abnormalities. These can be scored as chromosomal aberrations or micronuclei. Both forms of chromosomal damage are characterized by a defined dose-response relationship and a high stability of signal with time, provided that the exposed cells do not undergo postradiative division. These characteristics make chromosomal damage an ideal tool for the quantitative assessment of radiation damage, especially in human peripheral blood lymphocytes (PBL) that rarely undergo spontaneous cell division.

The level of chromosomal damage in PBL is routinely used as a biological dosimeter following human in vivo exposure. An exciting field of application is medical exposure to radiation. One of us has, in the past, tested the applicability of biological dosimetry in radiotherapy of thyroid cancer with I-131, brachytherapy of restenosis with P-32 and teleradiotherapy of mammary carcinoma.

Micronuclei in PBL of patients treated with I-131

Application of radioactive iodine-131 to the therapy of thyroid cancer can be associated with the induction and persistence of cytogenetic damages in healthy tissues in vivo. This study was performed in collaboration with the Military Hospital and the Institute of Nuclear Chemistry and technology, Warsaw, Poland. We aimed to evaluate the dose-effect relationship between the radiation doses received by patients treated with I-131 and the frequency of micronuclei in circulating lymphocytes of these patients. The purpose of the study was to verify the sensitivity of the micronuclei assay to detect a putative exposure to radioactive iodine.

Twenty-nine patients with various forms of thyroid cancer who were treated with 166 MBq to 6.5 GBq of I-131 were included in the present study. From these patients blood samples were collected on day 4 following administration of the radioiodine, lymphocytes were separated and scored for the number of cells with micronuclei per 1000 binucleated cells. As controls, lymphocytes obtained from 34 control healthy, sex- and age-matched donors who were not treated with I-131 were used. The results are shown in figure 1.

Although a significant difference in the frequency of micronuclei was observed between the I-131-treated and control subjects (figure 1 – insert), the weak dose-effect relationship indicates that the micronucleus assay is not sensitive enough to detect exposure to this radioisotope at doses used for the treatment of thyroid cancer.

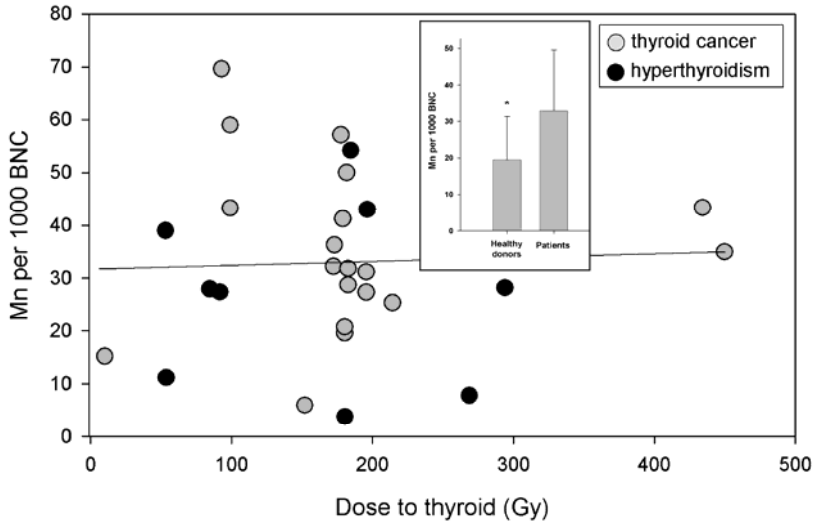

Figure 1. *Micronucleus (Mn) frequencies as a function of dose to the thyroid gland in lymphocytes of patients treated with I-131. Insert: average Mn frequencies in lymphocytes of patients and healthy control donors.*

Micronuclei in peripheral blood lymphocytes of patients treated for restenosis with 32P endovascular brachytherapy

Restenosis is a complete occlusion of the blood vessel leading to such complications as ischemia/ angina, myocardial infarction or death. It can be managed by endovascular brachytherapy with both gamma and beta sources. Endovascular brachytherapy is performed worldwide on several thousand cases per year. The gamma-emitter Ir-192 or the beta-emitters P-32 and Sr-90 are mainly used. The dose to the occluded endothelial wall is 20 Gy. Interestingly, no data exists with respect to the dose absorbed by the blood during the course of the treatment. The aim of the present investigation was to verify if the micronucleus test is suitable to detect the dose absorbed by lymphocytes in the course of endovascular brachytherapy with P-32. The study was performed in collaboration with the Institute of Nuclear Chemistry and technology, The Oncology Centre and the Institute of Cardiology, Warsaw, Poland.

Blood was drawn from 16 patients immediately before and one day after the treatment. Frequencies of micronuclei were assessed. In order to ensure that the micronuclei did not arise due to fluoroscopy or reperfusion, we analyzed lymphocytes of 16 control patients who underwent interventional cardiology with balloon angioplasty only.

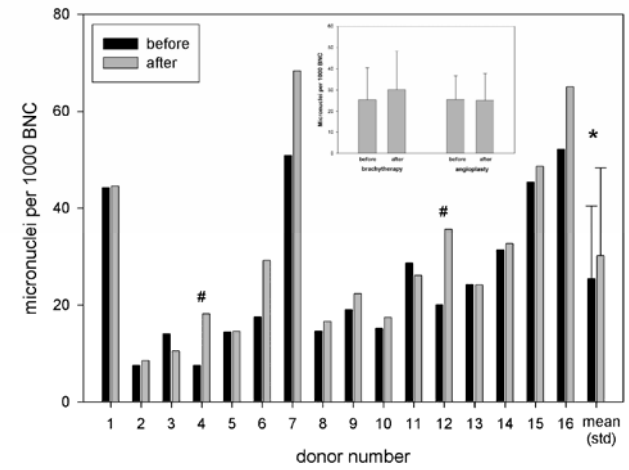

Figure 2. *Micronucleus frequencies in PBL of patients collected before and after brachytherapy. Insert: average frequencies in PBL of patients treated with brachytherapy and blood angioplasty.*

Enhanced frequencies of micronuclei were observed in lymphocytes of some donors following brachytherapy (figure 2). No correlation could be detected between the level of induced micronuclei and the absorbed dose. Also no effect of fluoroscopy or reperfusion was seen. Thus, although brachytherapy of restenosis with 32P leads to a weak enhancement of the micronucleus frequency in lymphocytes, the effect was not seen in all patients and the reason for this heterogeneous response remains to be elucidated.

Studies planned in the new laboratory of radiation biology at the Stockholm University

Recently, a new laboratory of radiation cytogenetics was set up at the GMT department of the Stockholm University. Here we plan to study the effects of exposure of PBL to mixed beams of radiation and to a changing dose-rate.

With respect to mixed beams the question is whether radiations of different quality (LET) act synergistically or additively when given together. Such exposure scenarios occur in BNCT, where patients are exposed to a combination of alpha and gamma radiation but also in places on Earth with a high natural level of radiation, in space and in aircrafts. The limited data published so far does not yield a clear answer about the effect of a combined exposure and the answer is important for proper assessment of the health risk of such exposure. A major drawback of the hitherto published studies is that cells were exposed successively to low and high LET radiation. No one has performed the experiments in such a way that the cells are exposed to both types of radiations simultaneously. This latter approach is the goal of the panned studies. Human peripheral blood lymphocytes will be exposed to alpha radiation, gamma radiation and to a combination of both sources. Chromosomal aberrations will be analysed as the endpoint. Presently the alpha source is being build. It can be expected that the results will yield new information about the interaction of high and low LET radiation.

In the environment people are frequently exposed to changing dose rates. The best example is aeroplane passengers during take-off and landing. During take-off the dose rate increases from the background value (of about 0.3 μSv/h) to about 5 μSv/h at the cruising elevation of 10 km. The situation is reversed during landing. Strikingly, the biological effects of exposing cells to changing dose rates have never been systematically analyzed although, given the frequent exposure of people to such irradiation scenarios, it is a very interesting and important problem from the point of view of radiation protection. We plan to study the effect of increasing and decreasing dose rate on the level of chromosomal aberration in PBL. Preliminary results indicated that the level of damage is lower in cells exposed to an increasing dose rate then when exposed to a decreasing dose-rate. The planned experiments will show if the effect is reproducible.

Studies of bystander effects in human artificial 3D tissue systems following charged particle microbeam irradiation

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The universality of the target theory of radiation-induced effects is challenged by observations of non-targeted effects such as bystander effects. Essential features of bystander effects are that they do not require direct nuclear exposure by radiation and they are particularly significant at low doses. Radiation effects at the tissue level under normal conditions prove that individual cells cannot be considered as isolated functional unit within most tissues of a multicellular organism. Experimental models, which maintain tissue-like intercellular cell signalling and three-dimensional (3D) structure, are essential for proper understanding of bystander effects. The main rationale for our research is that the bystander effect is likely to be natural phenomena which should be studied in an *in vivo* like multicellular system with preserved 3D tissue microarchitecture and microenvironment. This necessitates moving from *in vitro* cell culture systems to tissue-based systems. Our current work relates to human artificial epidermal (EpiDerm) and trachea-bronchial (AirWay). Charged particle microbeams are a powerful tool for investigating mechanisms of bystander effects. Microbeams are facilities that allow irradiation of individual cells or cell regions with a precise numbers of charged particles with micrometer precision. In our studies we were using counted alpha-particle, 3He and proton exposures. We demonstrated strong bystander induced apoptosis and premature differentiation after very low dose irradiation, which propagates up 1000 micrometers from irradiated spot. A better understanding of non-targeted effects may have important consequences for health risk assessment and, consequently, on radiation protection. 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. Further research is required to determine if these effects, typically measured *in vitro*, are applicable in tissue level, whole animals, and ultimately in humans.

1 POSTER

Non-targeted effects of ionising radiation (NOTE) – a new European Integrated project, 2006-2010

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The general objectives of the NOTE project are: (1) to investigate the mechanisms of non-targeted effects, in particular, bystander effects, genomic instability and adaptive response; (2) to investigate if and how non-targeted effects modulate the cancer risk in the low dose region, and whether they relate to protective or harmful functions; (3) to investigate if ionising radiation can cause non-cancer diseases or beneficial effects at low and intermediate doses; (4) to investigate individual susceptibility and other factors modifying non-targeted responses; (5) to assess the relevance of non-targeted effects for radiation protection and to set the scientific basis for a modern, more realistic, radiation safety system; (6) to contribute to the conceptualisation of a new paradigm in radiation biology that would cover both the classical direct (DNA-targeted) and non-targeted (indirect) effects. The NOTE brings together 19 major European and Canadian groups involved in the discovery, characterisation and mechanistic investigation of non-targeted effects of ionising radiation in cellular, tissue and animal models. The NOTE research activities are organised in six work packages. Four work packages (WPs 2-5) are problem-oriented, focussing on major questions relevant for the scientific basis of the system of radiation protection: WP2 Mechanisms of non-targeted effects, WP3 Non-cancer diseases, WP4 Factors modifying non-targeted responses, WP5 Modelling of non-targeted effects. The integration activities provided by WP6 strengthen the collaboration by supporting the access to infrastructures, mobility and training. WP7 provides dissemination and exploitation activities in the form of workshops and a public website. Managerial activities (WP1) ensure the organisation and structures for decision making, monitoring of progress, knowledge management and efficient flow of information and financing. Coordinator of the NOTE project is Prof. Sisko Salomaa, STUK - Radiation and Nuclear Safety Authority, Helsinki, Finland. The project is supported by the European Commission under the Euratom specific programme for research and training on nuclear energy, 6th Framework Programme. Please visit the project website http://www.note-ip.org to obtain more information or contact us by e-mail note@stuk.fi.

4 Natural Radiation and Radiation in Nature session

Disposal of Uranium from Drinking Water Filters

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Background

The recent EC Drinking Water Directive and Recommendations from the European Union, on uranium and other radionuclides in drinking water may lead to an increased use of various types of drinking water filters. Recent measurements on 700 samples from drinking water indicate a number of regions with increased uranium levels, such as around lake Siljan in mid-Sweden. The report is now in preparation and it can be assumed that these results may also lead to an increased interest in water filters.

To ensure that filters can be disposed of, preliminary calculations have been made on a hypothetical future stream of spent filters to a municipal waste disposal site.

It should be mentioned that, despite the recent concern for uranium, radon is still the main problem in connection with drinking water from a radiation protection point of view.

The finding of the recent measurements to be published shortly (March 2008) shows that there is no clear correlation between occurrence of radon and uranium.

Discarded water filters as a waste stream

The process using of water from wells is a practice that cannot easily be replaced, although it occurs very infrequent. The justification of the practice, i.e. of water extraction, is therefore seldom in question.

Accumulation of uranium in water filters is also a process unknown to most well-owners. This points to the need for societal support in the process of waste management. Another argument for this is that the alternative to waste disposal is an uncontrolled waste stream where the waste is deposited

The dose rate at the filter's surface was 2 μSv/h and the radium-226 concentration was around 20 kBq/kg in the dry filter sand, two times higher that the concentration in the BSS exemption criteria. The BSS exemption levels are relevant in Sweden since there is no clear distinction given in the Swedish radiation protection law and its ordinance regarding waste management. (Regulation of General exemption may be issued, however.)

Many alternative ways of storage come to mind for this old filter/waste, which offer an input to the discussion of dose criteria and about realistic choices involved.

Figure 1. *An old filter was used 10–15 years, and then dispached to the back yard.*

Scenario screening

Initially, a number of scenarios were considered, such as

1. Exposure from external gamma radiation to a private person near the kitchen sink (assuming a small filter below the sink).

2. Exposure from external gamma radiation to a private person from a filter placed near the bedroom.

3. Exposure from external gamma radiation to a private person from a filter placed in the cellar near a washing machine.

4. Internal exposure of a private person in connection with change, or handling of the filter for other reasons.

5. Internal exposure of an employee person in connection with change, or handling of a filter.

6. Exposure of personnel in the waste management chain from collection point to the final disposal on a municipal disposal site.

7. Exposure in the future as result of intrusion after the disposal site has degraded and the institutional control has failed

8. Exposure in the future as result of water consumption after the disposal site has degraded and the institutional control has failed, and a well has been drilled into the aquifer below the site.

It should be mentioned that filtering water in large amounts is a relatively new activity, and today only 2 of 50 wells tested in the County of Dalarna, used as an example in this presentation, were equipped with filters. The goal for the study is therefore mainly to give a basis for information to potential filter buyers and other actors about availability of a waste solution. The scenarios are therefore mainly related to potential future streams, rather than on-going practices.

For reasons of priority, scenarios involving private persons were therefore considered first. The County Administration Board, Municipalities and private companies can be reached by SSI in a more organized manner.

The external gamma exposure of a person in a bedroom near the filter was the limiting scenario. This corresponds to the social scenario where a family with limiting housing space makes an extra room available in the cellar (next to the waster installation) e.g. to a teenager.

The highest radium-226 concentration found in the project, 7 Bq/l, was used to make screening calculations, which showed that 0.1 mSv /year would not be exceeded.

For internal exposure, simple screening calculation was made using a simple proportionality between the content in a person, drinking unfiltered waste, vs. the content in a filter. The ingestion is assumed to be the result of less rigorous hygiene in connection with change of the filter.

The screening result was also below 0.1 mSv/year. For this scenario, somewhat more arbitrary assumption cannot be avoided, i.a. the assumed ingestion of 0,1% of the filters' radioactive content.

Still, all screening calculations were made using very high levels of radioactive material, representing the highest radioactive content in a well. That would represent about 1 in 1000. It would not be a problem in national heath considerations but in a number of individual cases it may still be exceeded, and in some cases unknown to the consumer.

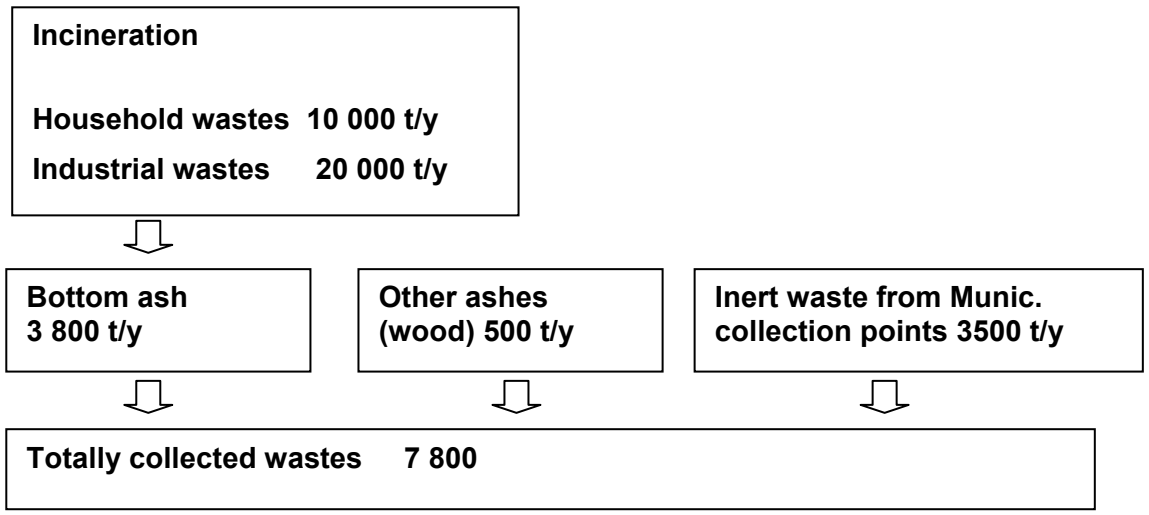
The waste stream to a disposal site

The main task to assist the public was to ensure that if filters for uranium were installed, they could be disposed of later. For this reason, potential waste streams of disused filters to a municipal disposal site were investigated.

The waste disposal site example

The example is made for the county Dalarna and it assume that all filters are deposited with their radioactive content on the disposal site in Borlänge. The streams are shown in Figure 1.

Waste streams (ton per year) to the disposal site in Borlänge.



It was assumed that 10 % of the 13 000 wells in Dalarna were equipped with filters, and no consideration given to back flush. Mean values of U (238 + 234) was assumed to be 1,4 Bq/l. The Ra-226 content was assumed to be 0,1 Bq/l.

The streams were supposed to flow for 100 years, the assumed age of the disposal site. Figure 2 gives the concentration in an aquifer below the site after the streams have ceased and the site closed.

The models are implemented in Ecolego Toolbox (Broed, 2007), which is a set of Simulink blocks to facilitate modelling of compartment based systems in the Simulink environment. The concentration of Th-230 is too low to be shown.

The maximum dose is in the order of 1 μSv/a achieved after several thousand years, indicating that filters should not represent a problem as a municipal waste stream. A final evaluation, including intrusion scenarios, there need to be consensus on scenarios for inhalation and ingestions as exposure routes.

The long time scales involved remind us all about the need and the value of information conservation in connection with environment-relevant information.

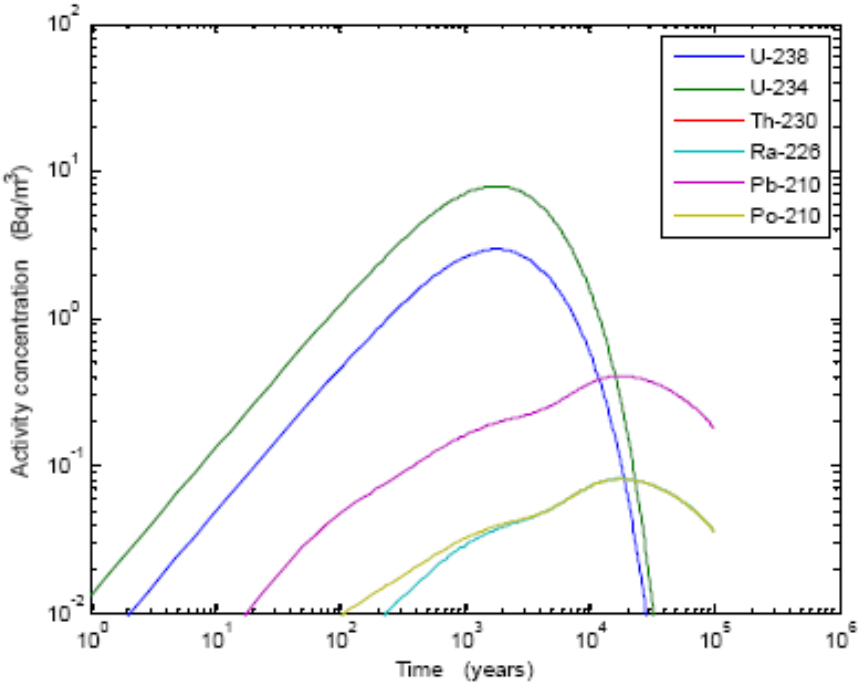


Figure 2 Activity concentration in well water as a function of time.

Nondestructive determination of gamma-active nuclide profiles in soil

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Radiactivity profile changes, in the same profile, can be followed nondestructively

A scintillator is set in tube or hole in soil (Fig. 1,3) and it measures the counting rate profile $c(z)$ of gamma-active nuclide photopeak. An inversion calculation, from this profile, is presented to determine the activity profile $q(z)$ of radioactive nuclide in soil. The stratified or plane symmetry of the nuclide distribution is assumed, so that q and c depend only on z. In inversion calculation the $q(z)$ is determined by iteration with the integral

$$c(z) = \int_0^\infty K(z', z) q(z') dz'$$

so long that fitting is the best. The kernel $K(z', z)$ is in [1].

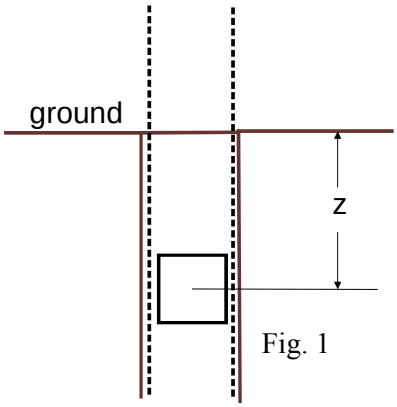

Fig. 1

^{137}Cs activity profiles, from the photo-peak (662 keV) profiles, are determined by using a 1" NaI(Tl) scintillator in tubes, which were in soils before Chernobyl, or in holes in peat. I started 1986 and the first result is in Fig. 3. In spring of 1987 STUK took my project concerning the cesium activities on swimming shores. I determined surface activities (the arrangement **a** of Fig. 2) in southern Finland , and also there in the tubes of the ground- and soil water sites of Hydrological office (today in Environmental Institute) I then used the arrangement **b** of the equipment in Fig. 2 for cesium profile determinations. ^{134}Cs was also in interests. Its disturbing effects in ^{137}Cs determinations were tried to eliminate. In profile calculations the density profile $\rho(z)$ ought to be considered. Its increase with water content can be determined by neutron moisture measurement or in other ways. In calculations I tried to consider the tube walls and snow on the ground. The Rayleigh scatterings (Fig. 4) have so small scattering angles that they cause no troubles. Polarization has effects in the both scatterings. The differential Klein-Nishina scattering cross section can be necessary to be multiplied by $S(x, Z)$ [2], where $x = \sin(\theta / 2) / \lambda$; λ is the initial photon wave length, θ scattering angle and Z atomic number. $S \in [0,1]$ and $S(0,Z) = 0$ for all elements. The compilation of $S(x,Z)$ in [3] I have used. The instrument in Fig. 2 is not so pulse height stabilized as should be. In future measurements the peat density instrument [4,5] I shall use, because it has the good stabilisation, from Nucletronics.

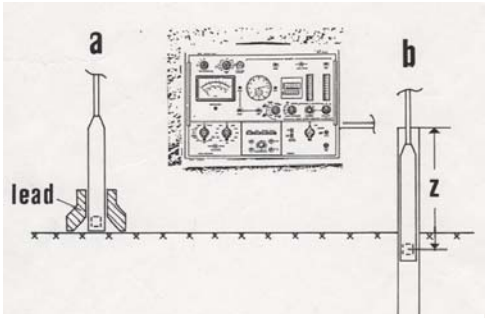

Figure 2. *Scintillator 1" NaI(Tl) and BASC scaler used in Cs surface activity (Bq/m^2) and soil profile measurement.*

In the first measurement and inversion calculation (Fig. 3) it was assumed ^{137}Cs layer with constant activity content. The forest soil profile in Tullinkangas in Lammi [1] I have measured most, but 1986-88 I started profile measurements also in Jämijärvi, Oripää, Orivesi and Elimäki in mineral soil, and in

Asikkala and Orimattila in peat. In the later calculations it was used the step wise constant approximation for $\rho(z)$ [1].

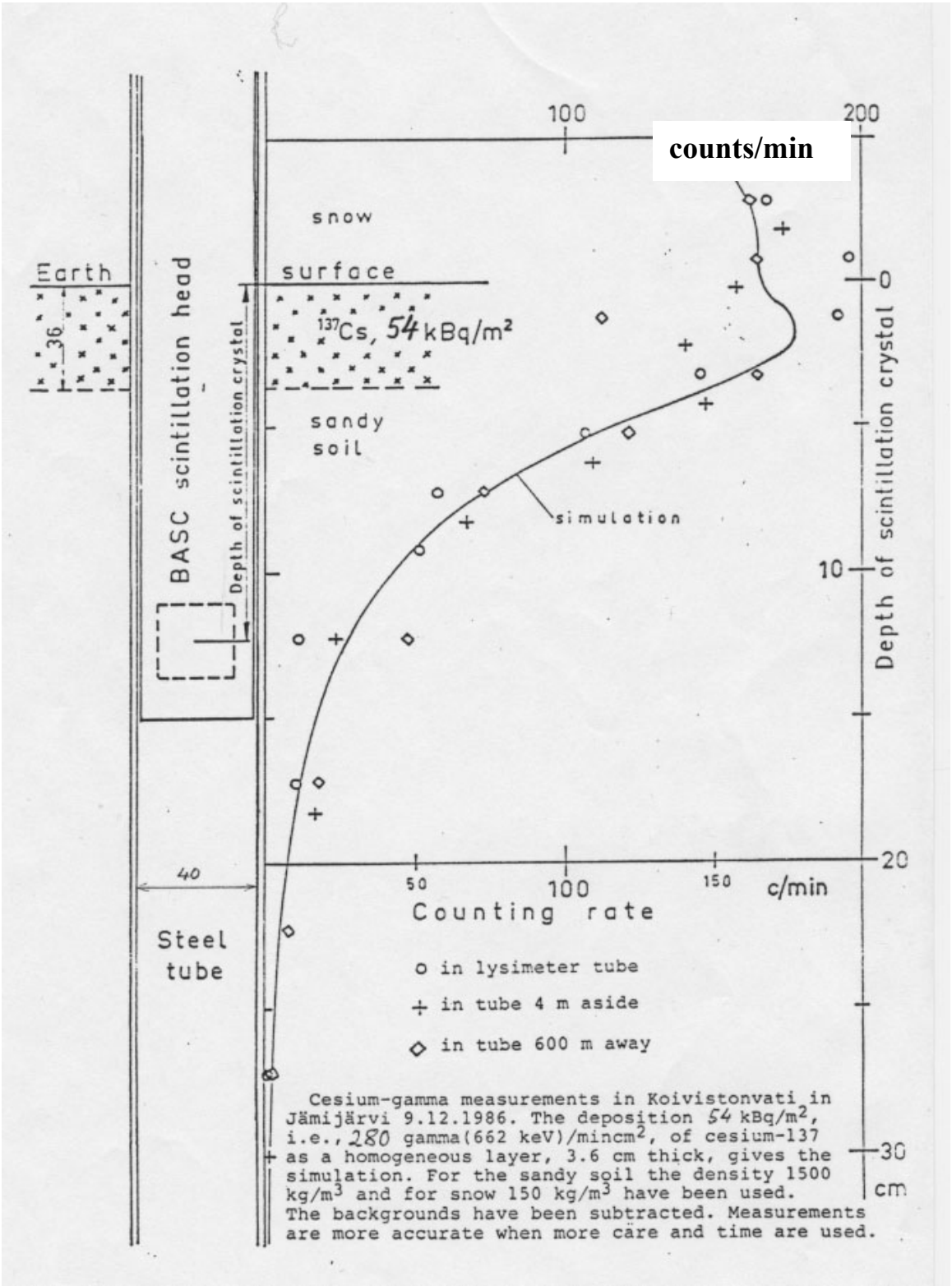

Figure 3. Cs-137 layer depth and activity determination with inversion calculation from the counting rates in 3 tubes in Koivistonvati, Finland, in 1986.

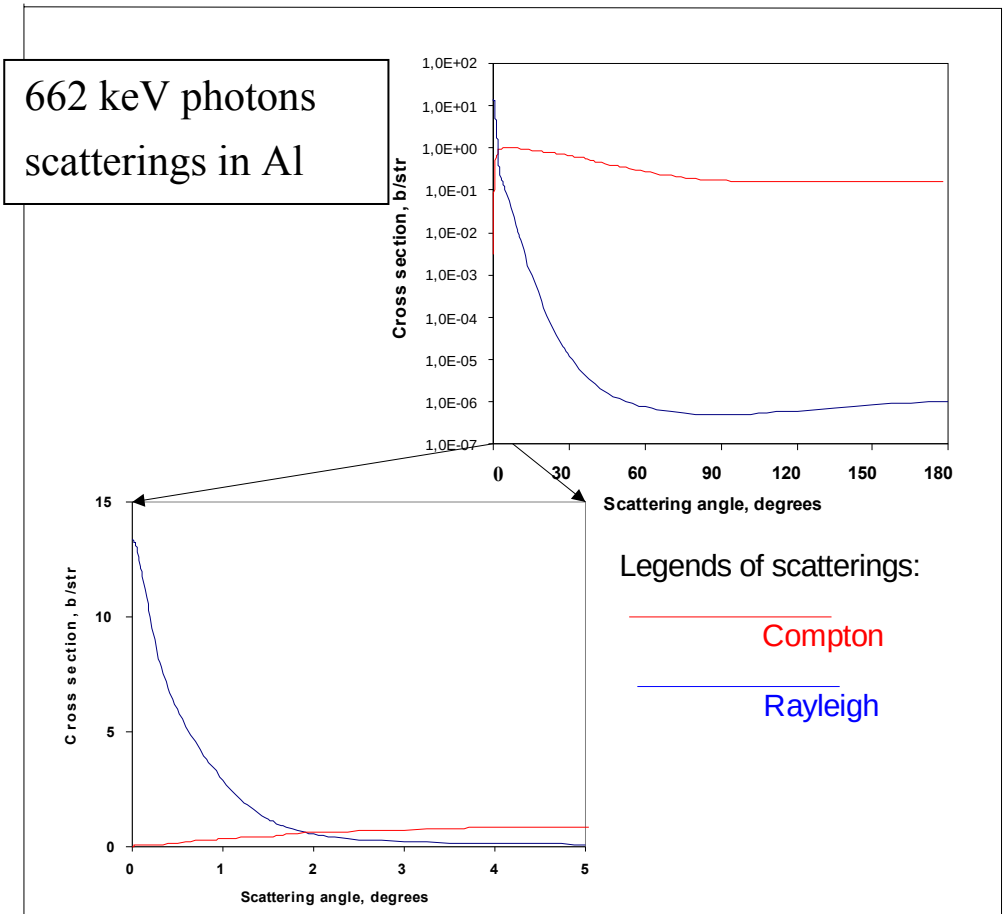

Figure 4. *Compton and Rayleigh cross sections on different scattering angles.*

After 1988 only one profile has been repeated in mineral soil site, and only 2 times [1], and once in peat. In the mineral soil, in Lammi in southern Finland, the stabilization of the ^{137}Cs profile has been observed [1], site B in Fig. 5. The site A is Koivistonvati of Fig. 3. Today I will verify the results in B and repeat all other profile determinations and make relevant new ones.

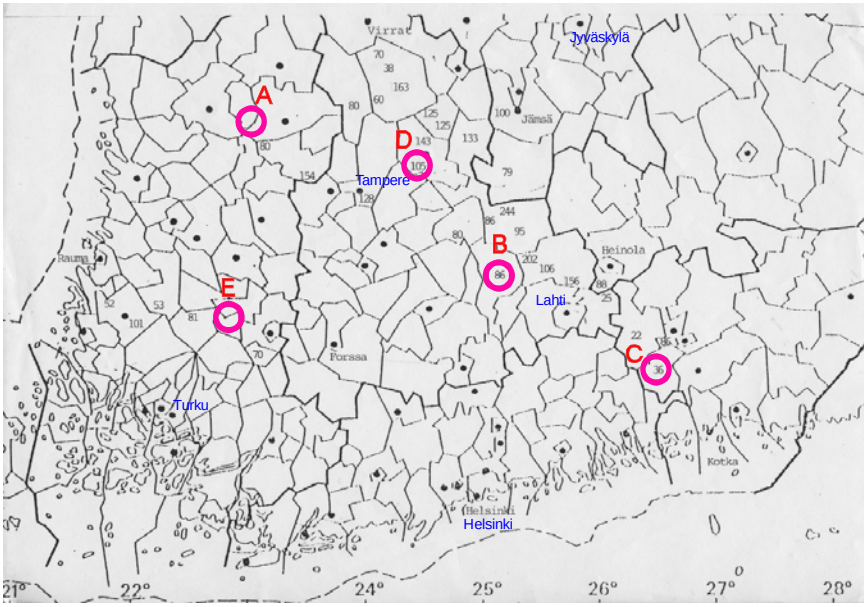

Figure 5. *The cesium profile determinations sites in red. The numerical values are cesium depositions in kBq/m^2 determined by the arrangement* ***a*** *in Fig. 2.*

The ^{137}Cs profile information is also important for determination of its contribution to human dose. In 1987 remarkable drifting of surface sand with its ^{137}Cs were found. In a sand pit to south-east from A the deposition was 80 kBq/m^2, but the drifting was found to cause this be increased even 5..6 times higher, mentioned also in [6].

In my poster in [7] there were presented the activities at the bottom of a pool in Padasjoki.

Potassium is used as a fertilizer. To follow its profile by using the 1461 keV photons of ^{40}K, I think, is interesting. I found plenty of findings of potassium fertilizers in web-search.

Snow water equivalent (snow mass) can be accurately determined by using snow attenuation of the cesium radiation. When counting, at the same time, the 1461 keV photons of ^{40}K as well, it may then be possible to determine the surface soil moisture. [6,7,8].

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Deposition of ^{237}Np in Finland

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Abstract

^{237}Np originates in environment from atmospheric nuclear tests, nuclear fuel reprocessing and from single nuclear events, which all have released ^{237}Np, ^{241}Am (mother nuclide) or ^{241}Pu (grandparent nuclide) to environment. Np is more mobile than other actinides in environment having several, easily changeable valence states. Furthermore, in the distant future ^{237}Np will be the dominating transuranium isotope in the environment due to its long, 2.16 million years, half-life and constantly increasing inventory from both direct ^{237}Np emissions and production by the decay chain ^{241}Pu → ^{241}Am → ^{237}Np.

The aim of the work was to determine total amount of ^{237}Np, deposited from nuclear weapons testing in 1950-1960´s and the Chernobyl accident, in Finland. Regional distribution of ^{237}Np in Finland was also sought. Furthermore, two analytical methods were tested, one based on ^{235}Np-tracer and other ^{242}Pu-tracer in ^{237}Np analysis.

Samples utilized in this study were collected from peat bogs in Southern and Central Finland immediately after Chernobyl accident. Previously activity concentrations of Pu isotopes, ^{243}Am and ^{244}Cm have been determined from same samples [1,2].

Separation method included wet ashing with concentrated acids, co-precipitation with calcium oxalate and TEVA® column separation. Concentration of ^{237}Np in sample was determined by ICP-MS. Recovery was determined either by measuring activity of low-energy photons/x-rays from ^{235}Np with Quantulus 1220 or concentration of ^{242}Pu with ICP-MS.

Activity concentration of ^{237}Np in peat at different bogs, total ^{237}Np deposition in Finland and differences between two analytical methods will be presented in more detail.

Introduction

^{237}Np originates in environment from atmospheric nuclear tests, nuclear fuel reprocessing and from single nuclear events, which all have released ^{237}Np, ^{241}Am (mother nuclide) or ^{241}Pu (grandparent nuclide) to environment. Np is more mobile than other actinides in environment having several, easily changeable valence states. Furthermore, in the distant future ^{237}Np will be the dominating transuranium isotope in the environment due to its long, 2.14 million years, half-life and constantly increasing inventory from both direct ^{237}Np emissions and production by the decay chain ^{241}Pu → ^{241}Am → ^{237}Np. There is no uniform opinion about possible enrichment of ^{237}Np in marine and terrestrial food chains, and more studies are needed to get a reliable view about behavior of Np in different ecosystems.

In this work, a geographical distribution of ^{237}Np deposition in Finland was studied and contributions from the nuclear weapons testing in 1950-1960´s and the Chernobyl accident were estimated. Samples utilized in this study were collected from peat bogs in Southern and Central Finland immediately after the Chernobyl accident (12.-14.5. 1986). Previously activity concentrations of Pu isotopes, ^{241}Am and

^{244}Cm and various gamma emitters have been determined from the same samples (Paatero et al. 1994, Salminen et al. 2005, Jantunen et al. 1991, Paatero et al. 2007).

Materials and methods

40-100 g of dried and homogenised peat was used for analysis. Separation method included ashing, wet ashing with concentrated acids, calcium oxalate co-precipitation and TEVA® column separation. Mass concentration of ^{237}Np in samples was determined by HR-ICP-MS (Micromass Plasma Trace 2). ^{235}Np tracer solution was added to each sample before wet ashing as a yield determinant. Chemical recovery was determined by measuring low-energy x-rays and Auger-electrons from ^{235}Np with the liquid scintillation counter Quantulus 1220. Counting efficiency of Quantulus 1220 was found to be 33% for L-shell transitions (12-21 keV) of ^{235}Np.

Results and conclusions

The median recovery of Np was 79 % with the chosen separation procedure. A large sample mass and high uranium concentration in many samples complicated the analysis and with many samples another TEVA® column separation was needed to reduce the U concentration to an acceptable level considering ICP-MS measurement of ^{237}Np. In recovery determination, the major problem was elevated background activity level of scintillation cocktail Opti Phase HiSafe 3. The analytical method needs further development to improve the separation efficiency of Np from U and to reduce background level at LSC by using some other scintillation cocktail with a lower background.

The results are presented in Table 1. Activity concentration of ^{237}Np in peat at different bogs was 0.48(±0.01)–3.84(±0.08) mBq/kg or 1.98(±0.05)–14.1(±0.03) mBq/m^2. For comparison, activity concentrations of $^{239+240}$Pu, ^{241}Am and ^{244}Cm in peats were 0.022(±0.003)–1.779(±0.087) Bq/kg (Reponen et al., 1993), 0.011(±0.002)–0.81(±0.03) Bq/kg and <0.0005–0.255(±0.017) Bq/kg (Salminen et al., 2005), respectively (reference date 1.5.1986). The activity concentration of ^{237}Np in peat is orders of magnitude lower compared to other transuranium nuclides. Lindahl et al. (2004) determined activity concentration of ^{237}Np to be 0.08(±0.01)–2.08(±0.17) mBq/kg in Swedish lichens and in the most heavily ^{137}Cs-contaminated areas, the fraction of Chernobyl-originated ^{237}Np was 5–30%. The results of Swedish lichens and Finnish peats are in concordance.

^{237}Np/^{239,240}Pu activity ratio in global nuclear test fallout and Chernobyl fallout can be calculated to be 0.0037 and 0.000125, respectively (Beasley et al. (1998), UNSCEAR (2000)). In our samples the ratio was 0.00145(±0.00009)–0.00286(±0.00010). The Chernobyl-originated ^{237}Np fraction of the total ^{237}Np deposition in peat samples was 0.5–13% (see Table 1). The activity of Chernobyl-derived ^{237}Np in peats is presented in Figure 1. The peat bogs investigated were located at a relatively narrow sector from the southwestern coast towards northeast, where the highest activities of Chernobyl-derived transuranium nuclides have been previously observed (Paatero et al., 1994, Salminen et al., 2005). Compared to other transuranium nuclides, the fraction of Chernobyl-derived deposition of ^{237}Np was low. 0–100% of ^{239,240}Pu, 10–99% of ^{241}Pu and 0.9–100% of ^{241}Am in peat samples were deposited from the Chernobyl accident fallout, the highest values are explained with skimming off varying amounts of old nuclear test fallout during peat production (Paatero et al., 1994, Reponen et al., 1993, Salminen et al., 2005).

Table 1. *Activity concentration of ^{237}Np and fraction of Chernobyl-derived ^{237}Np (%) in peat samples in Finland.*

Peat bog	Location	A ^{237}Np (mBq/kg)	Chernobyl-Np/total Np (%)
21 Löyniönsuo, Hankasalmi	62.2 °N, 26.3 °E	0.73±0.02	0.5
11 Viherperä, Kankaanpää	61.7 °N, 22.8 °E	0.48±0.01	13.3
95 Korpisalonneva, Vimpeli	63.1 °N, 24 °E	1.97±0.04	0.6
144 Kulvesuo, Rautavaara	63.5 °N, 27.6 °E	3.37±0.08	5.4
148 Kumpusensuo, Pielavesi	63 °N, 26.8 °E	3.84±0.08	2.3
99 Korvaneva, Jalasjärvi	62.3 °N, 22.9 °E	1.81±0.04	4.6

It is estimated that the activity of ^{237}Np in Finland from the Chernobyl accident was $1.1 \cdot 10^{7}$ Bq, based on $1 \cdot 10^{11}$ Bq of ^{239,240}Pu from Chernobyl to Finland (Reponen et al., 1993) and inventories of ^{237}Np and ^{239,240}Pu in the reactor core at the time of the accident (UNSCEAR, 2000). From global fallout, $7.4 \cdot 10^{10}$ Bq of ^{237}Np has been fallen to Finland (Hardy et al., 1973, UNSCEAR, 2000, Beasley et al., 1998). It can be concluded that ^{237}Np in peat in Finland originates mainly from global nuclear test fallout, and concentration of ^{237}Np, both Chernobyl-derived and older global fallout, varies depending on location.

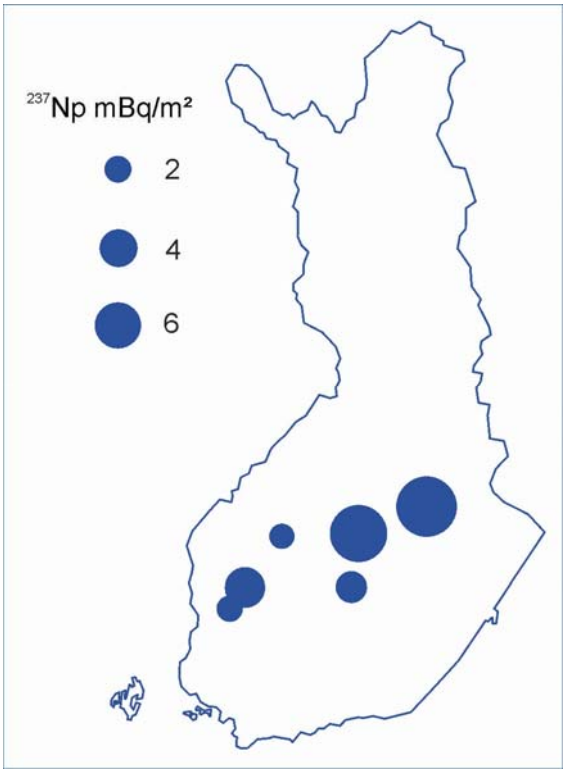

Figure 1. *Deposition of Chernobyl-derived ^{237}Np in peats in Finland.*

Acknowledgement

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Radioactivity in produced water from Norwegian oil and gas installations - concentrations, bioavailability and doses to marine biota

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Large amount of produced water, containing elevated levels of radionuclides (mainly ^{226}Ra and ^{228}Ra), is discharged to the sea in connection with oil and gas production on the Norwegian Continental Shelf. Until now, no study has assessed the potential radiological effects on marine biota in connection with radionuclide discharges to the North Sea.

The main objective of the project is to establish radiological safe discharge limits for radium, lead and polonium associated with other components in produced water from oil and gas installations on the Norwegian continental shelf. We here report the methodology developed and the overall results obtained.

Our results show that presence of added chemicals like scale inhibitors in the produced water have a marked influence on the speciation of radium, i.e. formation of radium and barium sulphates when produced water is mixed with seawater. The mobility of radium (and barium) is much larger than anticipated. Radium's behaviour in the sea has been modelled using the DREAM model.

Also, we have showed that the bioavailability of radium may be increased due to presence of scale inhibitors. Juvenile cod has been exposed to radium through food and from water. The uptake in different organs, i.e. skin, gills, liver, blood, and kidneys, show large variability, but is modest in all organs.

Reference organisms for the North Sea area have been chosen for calculation of absorbed dose to biota.

Dose rate from gamma radiation in dwellings – a modelling approach

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Abstract

Simulations of gamma dose rate in the indoor environment were made using the software MicroShield 6.20 (Grove Software Inc., Lynchburg, VA, USA). The sources included were building material (concrete, as well as light weight concrete) and indoor air. When simulating a detached quadratic standard room the dose point was at the centre of the room, with a shortest point-to-source distance of 125 cm. The quadratic room was modelled by wall slabs 400×250×20 cm^3 and slabs representing the floor/ceiling with dimensions 400×400×20 cm^3. The activity per unit mass of ^{238}U in concrete was chosen to 100 Bq/kg and light weight concrete was assigned an activity per unit mass of 1000 Bq/kg. With these assumptions, a dose rate of 78.6 nGy h^{-1} was achieved for a concrete building and 434 nGy h^{-1} for a light weight concrete construction containing alum shale.

For concrete walls the dose rate from radionuclides in indoor air (assuming equilibrium between ^{222}Rn and the decay products, ^{218}Po, ^{214}Pb, ^{214}Bi and ^{214}Po, at a radon concentration of 50 Bq/m^3) was calculated to 0.30 nGy h^{-1}, while the corresponding figure for a room built from light weight concrete (assuming 800 Bq/m^3) was 4.8 nGy h^{-1}.

Different room geometries were modelled and the contribution from the building material in adjacent and remote rooms was calculated. The dose rate in the modelled room was studied by varying its location in the building (single storey to 5 storey building). The largest relative difference was found between a corner room on the 1st or 5th floor and a centre room on the 3rd floor. The increase in dose rate was 37 % for alum shale concrete and 4.1 % for regular concrete, respectively. Thus, the contribution from adjacent rooms ought to be considered when predicting the dose rate in light weight concrete constructions.

Introduction

In many countries the population spends a substantial part of their time indoors, which makes the exposure to ionising radiation in the indoor environment important when considering the total radiation dose to the population. In Sweden, it has been assumed that the population spends 90 % of their time indoors (Andersson et al., 2007). The indoor radiation environment is to a high extent determined by the composition of the building materials, which may contain considerable amounts of naturally occurring radionuclides, but is also affected by cosmic and terrestrial radiation, as well as by radiation from fallout radionuclides. Although concrete and brick walls provide substantial shielding from gamma sources outside the building (Finck, 1992), the radioactive elements present in these building materials may often result in higher dose rates indoors than outdoors (UNSCEAR, 2000).

To model the external dose rate in a dwelling due to the building materials, a number of methods have been presented in the literature: *e.g.* Máduar and Hiromoto (2004) found that the most important factors affecting the dose rate were the thickness and the density of the wall, when the activity concentration in the building material was kept constant; Tsutsumi *et al*. (2001) found that predictions made by a spherical room model assuming walls of thickness larger than 25 cm agreed with measurements in a quadratic room with the same activity concentration, wall thickness and total wall area.

The aim of this project was to: (*i*) model the dose rate in a standard room, taking into account the gamma radiation emitted from the walls, floor and ceiling in the room as well as the gamma radiation from the air volume inside the room; and (*ii*) consider also the contribution from adjacent rooms using these room models to estimate the dose rate in a certain room, given the position of the room in the building.

Material and methods

Using the software MicroShield 6.20 (Grove Software Inc., Lynchburg, VA, USA) the calculation of photon fluence from a slab source to a "dose point" is made by assuming a uniform distribution of radionuclides in a slab of homogeneous composition and summing the contribution from a large number of point sources within the slab. The materials between the point source and the dose point absorb and scatter the photons and the attenuation and build-up is calculated by the software, *i.e.* total (with build-up effects included) as well as primary photon fluence is calculated. In this work, only total fluence has been used.

The radiation source was simulated assuming a given activity per unit mass of the ^{238}U-series elements in equilibrium. Since calculating the fluence from all gamma energies is very time consuming the photons were grouped into 25 groups, ranging between 15 keV and 2.0 MeV, and photons belonging to a given group was then handled by the software as having the mean energy of the group. The activity per unit mass of the building materials, as well as the type of building material, was chosen to resemble a common dwelling in Sweden. Hence the two building materials chosen were concrete (density 2350 kg/m^3) and light weight concrete (800 kg/m^3). The activity of ^{238}U per unit mass in concrete was chosen to 100 Bq/kg, which is just above the mean for Swedish bedrock (Stranden, 1988; Sundevall, 2002), giving an activity per unit volume of 0.235 Bq/cm^3. The light weight concrete was assigned an activity per unit mass of 1.00 kBq/kg (0.800 Bq/cm^3), corresponding to the mean ^{238}U activity for this material when containing alum shale rock (Stranden, 1988; Sundevall, 2002). Only concrete is available as a default material in MicroShield, (density 2350 kg/m^3) and when simulations were made with light weight concrete, the default concrete material was used, but with the lower density.

The quadratic room was modelled by wall slabs (400×250×20 cm^3) and slabs representing the floor/ceiling (400×400×20 cm^3). The dose point was set in the middle of the room, with a shortest point-to-source distance of 125 cm and the total dose rate inside a detached room was modelled by summing the dose rate contributions from all six delimiting slabs. During the integration process, each slab segment is divided in 64 000 smaller segments (40 slices in each spatial dimension) and the photon fluence at the dose point was thus calculated from 384 000 point sources within the slabs. A higher resolution was tested (80 slices in each spatial dimension), but no difference could be seen.

Due to restrictions in the software the contribution from gamma emitting radionuclides in indoor air was simulated using a cylindrical room model (instead of a quadratic), assuming equilibrium between ^{222}Rn and the decay products (^{218}Po, ^{214}Pb, ^{214}Bi and ^{214}Po) at a concentration of 50 Bq/m^3 (Stranden, 1988). In this room model, the inner radius was set to 235.9 cm in order to keep the inner slab area and distance from floor to ceiling constant (72 m^2 and 250 cm, respectively). In the light weight concrete configuration, the specific activity for ^{222}Rn was set to 800 Bq/m^3 air since activity concentrations exceeding this have been found in many light weight concrete buildings with an insufficient ventilation rate (Mäkeläinen *et al*., 2001; Sundevall, 2002; SSI, 2005).

The quadratic room model was used to create different macro structures, in order to compare the dose rates in rooms located in different parts of a block of flats. House types from 1 to 5 floors were considered where the ground floor was considered equivalent with the top floor since the outdoor environment was excluded. Three different horizontal positions were considered: corner, semi-centric

and central position (Figure 1). With this modelling approach, buildings with more than 5 floors will be equivalent to a 5-storey building regarding the two bottom floors and the two top floors. All the other floors will be equivalent to the 3rd floor in the 5-storey building model. The reason for this is that the influence from very remote rooms was neglected for both material types. In order to simplify and make the model more general, the thickness of the building material was kept constant for all boundaries (20 cm) and because of limitations in MicroShield, only shields parallel with the source slabs could be included. However, in reality, the oblique angles will increase the effective slab thickness and lead to an overestimation of the calculated dose rate.

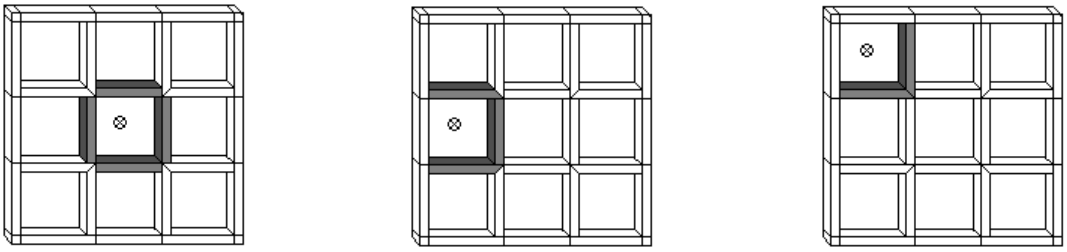

Figure 1. *The quadratic room shown from above in the central, semi-centric and corner positions, respectively. The shaded walls are delimiting the room from the rest of the rooms in the same plane. These walls, together with the ceiling and the floor of the room, were considered as primary shields against the photon fluence from the other rooms in the building. The rest of the delimiting walls, ceilings and, floors in the building were considered as sources and secondary shields. The dose points were located at the centre of the room and all walls had a thickness of 20 cm.*

Results

The dose rate increases with increasing thickness of building material until saturation is reached and 90 % (54 %) of the saturated dose rate was reached using a thickness of 20 cm concrete (light weight concrete) in the detached quadratic room. When simulating this room, a dose rate of 78.6 nGy h^{-1} was achieved for concrete and 434 nGy h^{-1} was found for light weight concrete containing alum shale.

The dose rate based on simulated total fluence from radionuclides distributed in the indoor air was calculated to 0.30 nGy h^{-1} (4.8 nGy h^{-1}) for a room with concrete (light weight concrete) walls.

The influence of building material in adjacent rooms is shown in Figure 2, and could be compared to the dose rate in a detached quadratic standard room that was found to be 78.6 nGy h^{-1}. The figure only shows the dose rates for a 5-storey building since the results were very similar for buildings with fewer floors. Figure 3 shows the increase in dose rate for different room positions in a concrete building with 5 floors relative to the dose rate in a corner room at the 1st (or 5th) floor. Corresponding simulations for a building made of light weight concrete are shown in figure 4 – 6. Contrary to the concrete case, we also found a considerable difference in dose rates depending on the height of the building. Therefore, also Figure 4 showing the dose rates in a 1-, 2- and 3-storey building is included. The dose rates in Figure 4 and Figure 5 could be compared to the dose rate in a detached quadratic room of light weight concrete, which was 434 nGy h^{-1}.

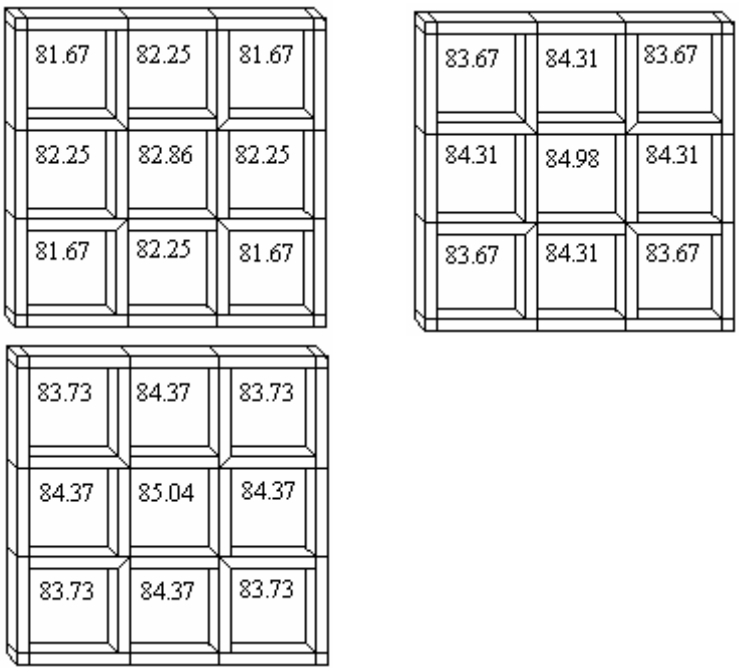

Figure 2 *Dose rates in nGy h^{-1} for different locations in a 5-storey building when using 20 cm thick concrete in all delimiting material slabs. Left panel: 1st or 5th floor; middle: 2nd or 4th floor; right: 3rd floor.*

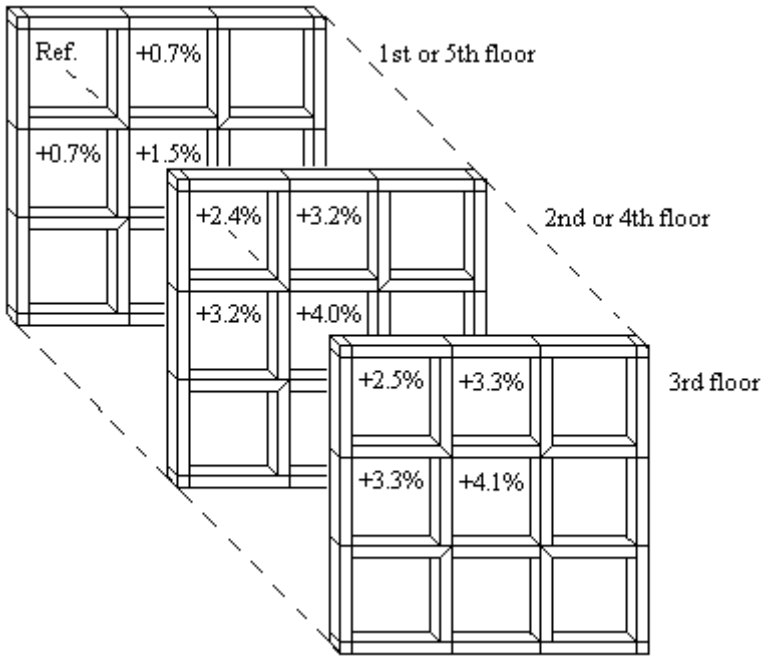

Figure 3 *Relative increase in dose rate for different room positions in a 5-storey concrete building, as compared with the dose rate in the corner room on the 1st or 5th floor.*

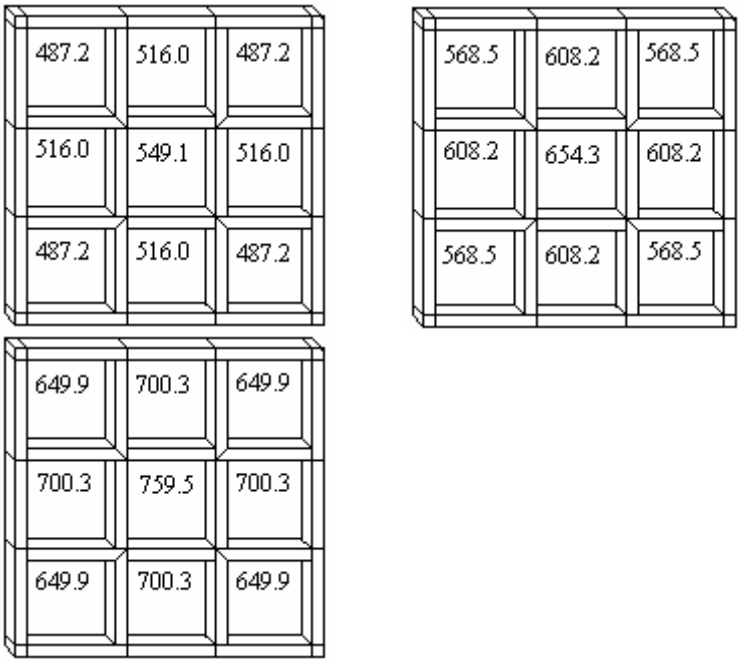

Figure 4 *Dose rates in nGy h^{-1} when using 20 cm thick light weight concrete in all delimiting material slabs. Left panel: 1-storey building; middle: 1st or 2nd floor in a 2-storey building; right: 2nd floor in a 3-storey building (the dose rate for the 1st and 3rd floor has been considered equal to the dose rate for the 1st and 5th floor in a 5-storey building (Figure 5) due to exclusion of the contribution from very remote floors).*

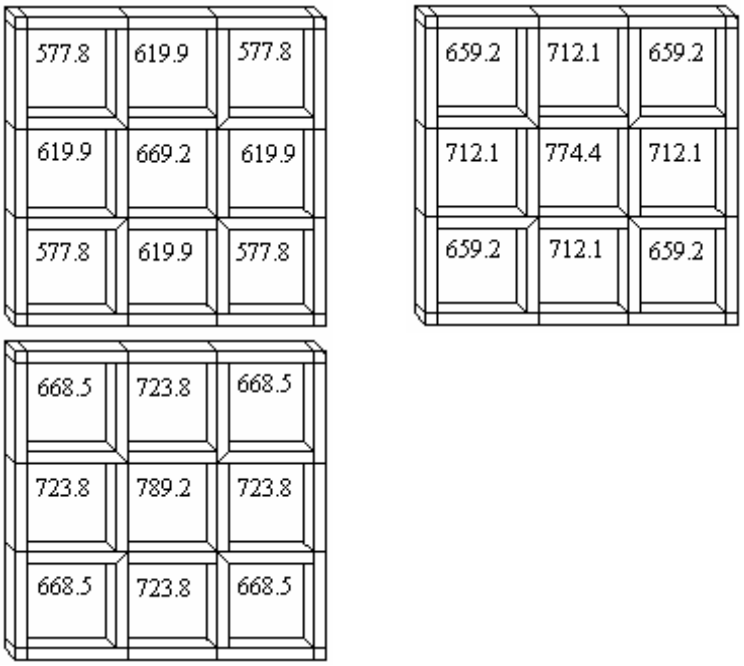

Figure 5 *Dose rates in nGy h^{-1} for different locations in a 5-storey building when using 20 cm thick light weight concrete in all delimiting material slabs. Left panel: 1st or 5th floor; middle: 2nd or 4th floor; right: 3rd floor.*

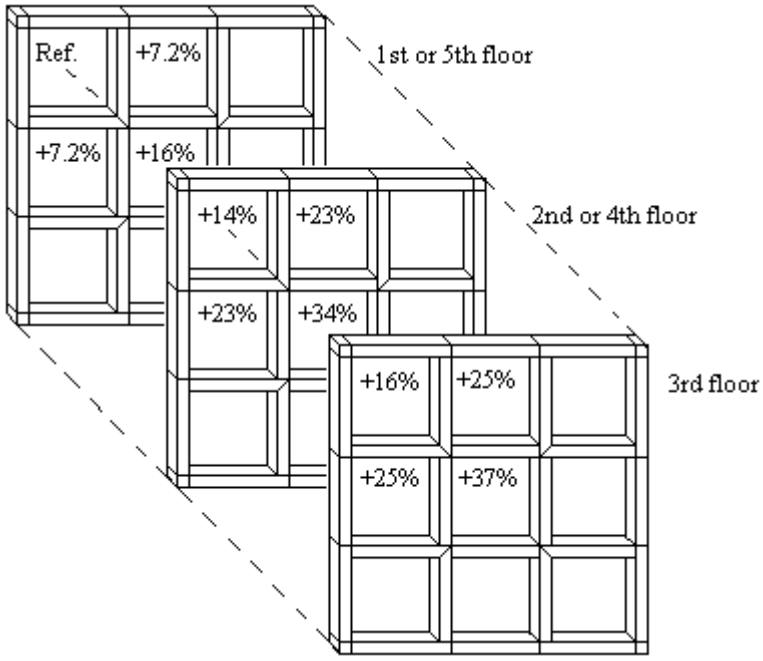

Figure 6 *Relative increase in dose rate for different room positions in a 5-storey light weight concrete building, as compared with the dose rate in the corner room on the 1st or 5th floor.*

Discussion and conclusions

No reinforcement bars, pipes or radiators were considered in the calculations and the influence of photon scattering from such objects, as well as shielding effects from furniture, might be significant in some parts of the room. In order to increase the accuracy of the calculations, Monte Carlo simulations could be used. The calculated dose rates for the concrete buildings were, however, close to indoor dose rates reported by Erlandsson & Isaksson (2005), who found an average outdoor dose rate from gamma radiation and cosmic radiation similar to the average indoor dose rate from gamma radiation, both close to 100 $nSvh^{-1}$.

The result from the quadratic room model for detached rooms showed that for a thickness of 20 cm concrete, 90 % of the saturated dose rate was reached. The corresponding result for light weight concrete was only 54 %, indicating that the macro structure would have a much greater relative influence on the dose rate in the latter case, when normal wall thicknesses are being considered. The difference in saturation thickness can be explained by the difference in density of the two otherwise similar materials.

Some interesting differences were found considering the dose rate in a 5-storey building. The largest relative difference was found between a corner room on the 1st or 5th floor and a centre room on the 3rd floor: 37 % for light weight concrete and 4.1 % for regular concrete, respectively. When moving

horizontally, from a corner to a centre room on the 3rd floor, the increase in dose rate was 18 % for light weight concrete and 1.6 % for regular concrete. Moving vertically from a semi-centric room on the 1st floor to a semi-centric room on the 3rd floor, the increase was 17 % for light weight concrete and 2.6 % for regular concrete. These results show that the expected annual dose to persons living in the same house can vary significantly in alum shale buildings, depending on the location of the apartment.

Another important conclusion from this project was that – when predicting the dose rate in a room with structures of low-density concrete – the dose rate from adjacent rooms should always be considered. Otherwise the predictions will underestimate the actual dose rate. This is of course even more important if the cement has been aggregated with alum shale or similar minerals with a high content of radionuclides. Considering a centre room at the 3rd floor in a 5-storey building the dose rate in a light weight concrete house (790 $nSvh^{-1}$) is nearly ten times the dose rate in a concrete building (85 $nSvh^{-1}$).

The relative contribution from the air inside the room was very low: 0.34 % for concrete and 0.65 % for light weight concrete, given the assumptions made about the activity per unit volume in the air. Thus the external dose is negligible compared to the internal dose due to inhaled radon daughters.

Future studies could consider other building materials and also include other radionuclides than those in the ^{238}U-series. By selecting real buildings similar to those modelled in this work, measured dose rates could be compared to the results predicted by the macro structural model.

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Radon i Norge – Status og strategi

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Statens strålevern, Norge

Abstract

Sammen med Sverige og Finland er Norge et av de landende i verden med høyest forekomst av radon i inneluft. Radon er den strålekilden som gir de største stråledosene til befolkningen i Norge. Internasjonale sammenligningsstudier har de siste årene styrket vår viten om sammenhengen mellom radoneksponering og lungekreft, og det er beregnet at ca 280 mennesker i Norge dør på grunn av lungekreft som skyldes radon hvert år. Statens strålevern vil derfor fortsette arbeidet med kartlegginger, og oppfordre privatpersoner til å gjøre målinger, samt stimulere til å gjennomføre tiltak. Radon er imidlertid ikke et problem bare i hjemmet, og Statens strålevern vil derfor arbeide for å få fokus på radonproblematikken på arbeidsplassen, i skoler og barnehager og i offentlige bygninger.

En interdepartemental arbeidsgruppe ledet av Helse - og omsorgsdepartementet arbeider for tiden med radonproblematikken i full bredde. Statens strålevern deltar i denne gruppen, som skal ferdigstille sin rapport høsten 2008. I denne presentasjonen vil status for radonarbeidet i Norge presenteres, og hvilke problemstillinger som tas med inn i arbeidsgruppen vil skisseres.

New indoor radon mitigation guides in Finland

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Abstract

Radiation and Nuclear Safety Authority - STUK and the Ministry of Environment have published in spring 2008 two new Finnish indoor radon mitigation guides. Both guides are revisions from previous 1990's issues. The first guide published by STUK gives basic information about all mitigation methods used in Finland. The most efficient method are sub-slab depressurization (SSD) and radon well, typical radon concentration reduction factors being 70-90% for both methods. The guide gives detailed information with many examples for SSD and radon well design and implementation. Sealing entry routes resulted normally in reduction factors of 10 - 40 %. Complete sealing may be requiring and expensive. Improving ventilation resulted only in few cases a reduction factor higher than 50%. Installing of fresh air vents in some cases decreases the depressure level of the house and the inflow of radon bearing soil air. Installation of SSD , radon well or fresh air vents are the most effective measures in blocks of flats. The guide handles briefly also radon mitigation in workplaces and big buildings. The second guide published by the Ministry of Environment is a detailed guide for design and implementation of a SSD in low rise residential houses. Practical information choosing the location of suction pit, implementation of suction pit and system piping has been given. The STUK-guide will be published also in Swedish.

Introduction

Finland belongs to the countries of high indoor radon concentrations. Cool climate, long heating season with no long term airing through windows, building soils with high air permeability and foundation structures promoting flow of radon bearing air from soil to indoor spaces are the main reasons to high indoor radon concentrations. Approximately 50.000 dwellings, 3% of all dwellings, exceed the action limit of 400 Bq/m^3. Most of these are low rise residential building. Similar problems are found in flats with floor slab in ground contact in the lowest floor.

The first indoor radon mitigation studies were carried out in mid 1980's. These studies resulted in first mitigation reports which gave basic information of active sub-slab suction installations. Both STUK and the Ministry on Environment published mitigation guides in 1990's. The STUK-guide gave an overview of all methods and the results achieved. The guide of the Ministry focused on sub-slab depressurization (SSD), design and implementation.

Both guides have been revised in 2008 (Arvela and Reisbacka 2008, Ministry of Environment 2008). The ministry guide is a detailed guideline for design. The STUK guide refers to the ministry guide and gives key principles for design and implementation.

Results achieved through mitigation methods

The STUK-guide reports mitigation results in 400 dwellings based on detailed mitigation questionnaire sent to house owners in 2000-2001.

Sub-slab depressurization (SSD) (also called sub-slab suction, "radonsug" in Swedish) and radon well are the most efficient methods. Typical reduction factors for both methods are 70-90%. In difficult cases additional sealing work is needed in order to achieve a low radon concentration. The reduction factors for other non-active methods are clearly lower, Table 1.

SSD can be implemented through both floor slab and foundation wall. The ministry guide focuses on the implementation of SSD through floor slab. The STUK guide gives examples and guidance also for foundation wall installations. SSD's have been installed in many cases through foundation wall in terraced houses where the floor slab area is not large. The reduction factors are typically above 80%.

Installation of a preparatory radon piping has become increasingly common in houses build during last ten years. Activation of this piping through an exhaust fan has resulted in high reduction factors typically above 80%.

A radon well will be constructed outside of the house and the well sucks air from soil from a depth of 3-4 metres. The air flow created ventilates soil air to distances of up to 30 metres. This ventilation decreases the radon concentration of soil air below the house foundations efficiently. Radon well is effective only on soils where air permeability is high enough e.g. on gravel and in esker areas.

Radon reduction method based on ventilation reduce radon concentration either through increased ventilation or decreased depressure level. Reduction factor above 50% have been achieved only in the cases where the original air exchange rate has been defective or when the depressure level has been high. Typical reduction factors have been 10 - 40%. Increasing the operation time or power of mechanical ventilation, opening existing or installing new fresh air vents are typical measures. Installation of new fresh air vents does not result normally in reduction factors above 50%.

Sealing entry routes aims at reduction of leakage flows of radon bearing soil air into living spaces. Sealing may be very requiring. In many cases the results are qualified only when the entry routes have been sealed almost completely. Best results have been achieved in houses where the foundation wall is of cast concrete. Floor joints with foundation walls of porous light weigh concrete can't be sealed with normal sealing methods.

Table 1. *Radon reduction factors achieved through different mitigation methods. The results are based in a questionnaire study in 400 houses. Well designed and implemented mitigations result in reduction factors which are better than the typical reduction factors in the table.*

Mitigation method	**Typical radon reduction factors, %**	**Average radon reduction factor, %**
Sub-slab suction (SSD)	70 - 90	75
SSD through foundation wall	70 -90	80
Preparatory radon piping and activated exhaust fan	75 - 95	85
Radon well	75 - 95	80
Improving ventilation,natural ventilation	10 - 40	20
Improving ventilation,mechanical exhaust ventilation	0 - 40	20
Improving ventilation, mechanical supply and exhaust ventilation	20 - 50	30
New mechanical exhaust ventilation	10 - 40	25
New mechanical supply and exhaust ventilation	20 - 50	40
Sealing leakage routes	10 - 50	30
Crawl space ventilation	30 - 80	60
Improving cellar ventilation	10 - 50	30
Several methods	30 - 70	50

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   - Dimensioning of air flows
   - Improvement of the efficiency
7. Air exchange

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Both publications will be available on website www.stuk.fi

Radon resistant construction in Finland in 2007

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Abstract

A revised guideline for radon resistant construction was published in Finland in 2003. The emphasis is on radon prevention in case of the most common substructure, i.e. slab-on-ground. The most important measures are sealing with bitumen felt and installation of radon piping. In this study the realisation of the guideline was investigated with a questionnaire and on spot measurements. Leakage measurements by using hydrogen as a tracer were carried out in 16 dwellings. It appeared that the major radon entry points are lead throughs and minorly the non-seamed joints of bitumen felt strips, especially in corners. The activated radon piping appeared to be effective in 11 out of 14 cases, but in three dwellings the indoor radon concentration still exceed the reference level 200 Bq/m^3. These observations set tight requirements to improve the sealing. If in doubt, it is safer to choose crawl-space foundation.

Introduction

In Finland, the main reason to elevated indoor radon concentration is radon bearing soil air. To prevent the soil air entry to indoor air, the National Building Code (2004) requires radon-technical design and radon-resistant structures in new building in the whole country. The reference level is set to 200 Bq/m^3. As guideline, the Code refers to the 2003 Radon Prevention Guide for Radon Resistant Building (2003).

The preventive design has become more common during the new millennium. For example, the radon campaigns which communes and the Radiation and Nuclear Safety Authority organise together, have shown that in many areas the radon piping is installed in more than half of the new buildings, and also the sealing has become more common.

The Radon prevention guideline for radon resistant building

The Guideline (2003) contains the instructions to seal foundations and to install sub-slab radon piping. The previous version of the guideline is based on the use of elastic sealants. In this revised version, bitumen felt is used for sealing the joint of the slab-on-ground and the foundation wall and possible walls backing ground. Lead-throughs of e.g. water pipes and electric cables should be sealed air tightly. The walls backing soil made of light weight concrete blocks should be plastered. This decreases soil air penetration in the case sealing does not fully succeed. In Figure 1 two examples of sealing are introduced.

The schematic figure of sub-slab piping is introduced in Figure 2. The piping is installed under the slab, and an exhaust duct is connected with the piping. If the indoor radon level exceeds 200 Bq/m^3, the piping is actived by connecting an exhaust fan to the duct. The system depressurises the sub-slab volume, which should stop or reduce strongly soil air entry. The system typically also reduces the soil air radon consentration by ventilating sub-slab volume with fresh outdoor air. The flow of the fan

should not exceed reference value 0,05 l/s per floor square meter, and a damper or electrical regulator system for the fan is sometimes needed.

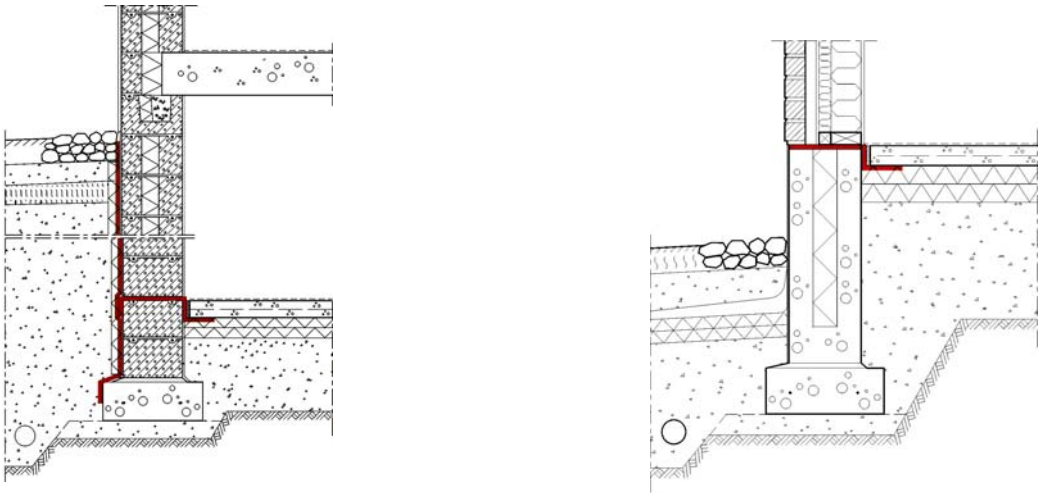

Figure 1. *On the left, the joint of the floor slab and concrete foundation wall is sealed with bitumen felt, marked with thick grey line. On the right, a light weight concrete wall is plastered on parts backing soil (both outside and inside, under the floor slab), to stop soil air entry. Bitumen felt, marked with thick grey line, is installed on outside parts backing soil, and to the joint of the floor slab and the foundation wall. In both cases, bitumen felt is installed directly towards the concrete cast floor slab. Figs. from RT 81–1079 (2003) /2/.*

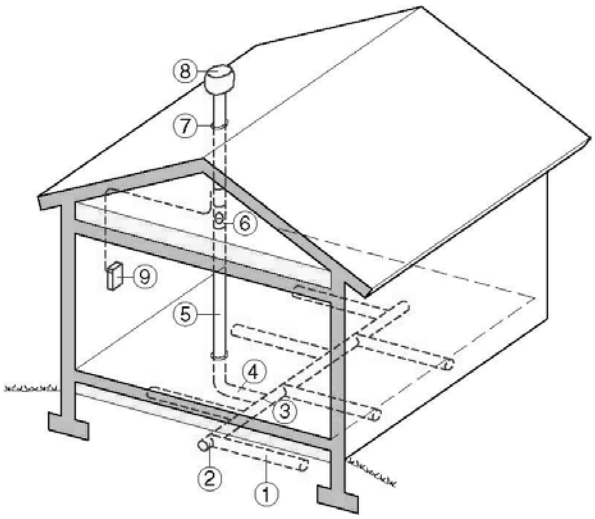

Figure 2. *A schematic figure on radon piping: (1) suction (radon) piping, (2) collector duct, (3) air removal point, (4) transmission duct, (5) exhaust duct, (6) damper, (7) roof follow-through, (8) exhaust fan and (9) possible electric regulator system for the fan. Fig. from RT 81–1079 (2003) /2/.*

Questionnaire

133 dwellings, built 2004 or later in Hyvinkää, Riihimäki, Kotka, Vantaa and Tampere region were selected to the study. In these dwellings, a radon measurement was ordered and carried out by the habitants during 2004 – 2006 by themselves. These measurements were used in the study. A questionnaire, with questions especially about sealing, installation of piping and exhaust fan status, was returned from 101 dwellings. A new radon measurement was offered to the dwellings, and it was

carried out during the winter 2006 –2007. The main purpose of the questionnaire was to select interesting dwellings for case studies.

Leakage measurements

The most important measurement to find radon entry points was carried out by using a tracer gas, containing 95 % of nitrogen and 5 % of hydrogen. The tracer was lead under the slab via a drilled hole in the foundation wall or, more often, by using a fan to pump air and the tracer gas into the radon piping. To find the entry points, a thorough investigation was made with a hydrogen sensor. The method is sensitive in finding the leakage points, but the flow of soil air cannot be measured. Some rough estimates of the severity of the leakage can be done by estimating the size of the leakage area and the spread of the detected tracer gas in the vicinity of the leakage point. This allows us to point the most significant leakages in clear cases. 16 dwellings were examined, and in 11 of them bitumen felt had been installed. The results are shown in the table 1. Some leakage points in the table can be less significant, as discussed above.

Table 1. *Leakage measurement in 11 dwellings with installed bitumen felt.*

entry point	number of dwellings with leakage
lead-throughs, significant leakages	10
non-seamed bitumen felt strips in corners	6
joints of slab and the bearing internal wall	5
close the doors	7
joints of slab and wall on straight wall parts	5
significant leakages on straight wall segments	1 (faulty installation of bitumen felt)
electric wall sockets	3
foundations of fire places	1

The leakages of lead-throughs are typically via protective pipings of electric cables or water pipes, especially the main water pipeline. On the other hand, the lead-throughs are sometimes easy to fix even afterwards. The joints of the bitumen felt strips are to be seamed with a gas burner, a hot air fan or with a specific kind of bitumen glue. According to discussions with constructors and dwellers, in most cases seaming was not done adequately. This was seen as point-like leakages in corners and also on the straight parts of the wall. Non-seamed strips are also the most probable reason for the leakages close the doors.

In one of the dwellings the bitumen felt was installed in a very different way not in line with the guideline. As a result, the joint of slab and wall leaked severely all around the dwelling. Generally, the most severe problem may appear if the bitumen felt is not installed at all. This was clearly seen in two reference dwellings, where the joint of slab and walls leaked all around. The unsealead lead-throughs are the next most severe penetration point of soil gas. The non-seamed bitumen felt strips may not result in significant leakages, but seaming is strongly advisable.

Sealing and indoor radon concentration

The number of dwellings with slab on ground sealed with bitumen felt was 23. The mean and median indoor radon concentration was 280 Bq/m^3 and 150 Bq/m^3, respectively, and the reference level 200 Bq/m^3 was exceeded in seven dwellings. During the measurements, the radon piping was not activated

with a fan. The result is in line with leakage measurements, where defects were found in many buildings. Unsealed lead-throughs complicate the interpretation of these results on sealing with bitumen felt.

Activated radon piping and indoor radon concentration

The radon piping was activated in 14 dwellings. The mean concentration before activation was 630 Bq/m^3 and median 430 Bq/m^3. After the activation the mean concentration was 130 Bq/m^3 and median 30 Bq/m^3. The indoor radon concentration decreased below the reference level 200 Bq/m^3 in 11 dwellings, but in three dwellings it was still too high. The average reduction in radon concentration was 82 % in all dwellings.

In two dwellings exceeding the reference level more thorough examinations were made. Mitigation was eventually successful, reducing the concentration below the reference level. Extra sealing work was needed in both cases. Exhaust fan flow exceeds in both cases the reference level. In unfavourable conditions this may lead into sub-slab temperature problems, and follow-up is needed by the dwellers. Most probably permeable sub-slab material made it too difficult for the exhaust fan to depressurise enough the sub-slab volume. Added to that, in one of the dwellings the exhaust was located much closer to the other end of a long-shaped semi-detached house, reducing much better the concentration in the other apartment.

Piping has problems with permeable soil

The guideline about radon piping is based on idea that the sub-slab and sub-foundation soil material is not very permeable. This indeed was the case still some years ago, but nowadays more permeable material is used. Depressurisation is not necessarily effective enough, and the air flow of the exhaust fan has to be strongly increased to compensate the problem. Exceeding the reference level of the flow may lead into sub-slab temperature problems. A more thorough study of the subject is needed.

All these observations emphasise the need for a careful sealing work, especially if the soil is permeable. If in doubt, one should consider a more radon resistant foundation than slab-on-ground, like crawl space.

Summary and Conclusions

The study showed clear defects in the sealing work of new buildings. The most severe problem is omission of bitumen felt, which often leads into significant leakages all over the joint of the slab and the walls. The leakage measurements demonstrated that lead throughs are not properly sealed, and the joints of the bitumen felt strips are not seamed. Lead throughs showed up to be the major entry points for soil air and radon.

If radon concentration exceeds the reference level 200 Bq/m^3 in spite of sealing, the radon piping is activated. Usually this is an effective prevention method, but in some dwellings the radon concentration was not reduced below the reference level without a thorough study and extra sealing work. The role of permeable sub-slab or soil material should be more thoroughly studied in the future.

These observations set high standard for the planning and realisation of the sealing work. If in doubt, one should consider choosing a more radon resistant foundation type, such as crawl space.

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Radon measurements in conventional Finnish workplaces during measurement seasons 2005-2007

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Abstract

In Finland the action level for the radon concentration in inhaled air is 400 Bq/m^3 in workplaces where people are working regularly (1600 hours per year). The radon concentration stands for the annual mean of the radon concentration during working time.

The responsible party, 408 employers, ordered and sent back 1630 radon measurements during seasons 2005-2007. Passive alpha track detectors of STUK were used for these mainly two-month measurements. The average of radon concentrations was 290 Bq/m^3 and the median 94 Bq/m^3. In 260 measuring places the radon concentration was more than 400 Bq/m^3, and in 94 places was more than 1000 Bq/m^3, the highest value being 13 300 Bq/m^3.

For 25 employers STUK verified the radon concentration during working time with Alpha Guard measuring equipments.

Introduction

In Finland the action level for the radon concentration in inhaled air is 400 Bq/m^3 in workplaces where people are working regularly (1600 hours per year). The radon concentration stands for the annual mean of the radon concentration during working time. The mean radon concentration may be higher than 400 Bq/m3, if the work is not regular. The action levels for the radon concentration shall also be applied to public buildings, such as schools, underground rooms intended for the public, and other comparable rooms.

The responsible party (employers) should measure the radon concentration in a conventional, aboveground workplace if the workplace is situated in a region where over 10 per cent of the annual mean values of the earlier radon concentration measurements in a work or residential environment exceed 400 Bq/m^3. STUK issues annually a separate list of municipality where radon measurements shall be made in workplaces and 2007 there was 88 municipalities on the list.

In addition to the above regions, the radon concentration is also to be measured in other parts of Finland, especially in workplaces situated on eskers or other gravel or sand formations which are readily permeable to air. STUK inspectors or local industrial safety authorities can also give instructions or orders for radon measurements. The radon concentration is also to be measured in all underground workplaces where work is permanently carried out. In industrial premises, it is usually sufficient to make 1 to 2 measurements. The measurement should be made in each separate building where people work. In case of office buildings, one measurement for each 200 m^2 is recommended.

Results of alpha track measurements and discussion

Between November 2005 and October 2007 employers made 483 orders including altogether 2083 alpha track measurements. Passive alpha track detectors of STUK were used for mainly two-month measurements. For this study it was possible to take results from 408 orders and 1630 radon measurements in working places.

The average of radon concentrations was 290 Bq/m^3, and the median 94 Bq/m^3.

16 % of measurements exceeded the action level 400 Bq/m^3, and 29 % exceeded 200 Bq/m^3.

Very high radon concentrations were measured in sub-basements of large buildings and in underground tunnels.

Table 1. *Statistics of radon measurements, classified by type of order*

Type of order	Number of orders	Number of radon measurements	Average Bq/m^3	Maximum Bq/m^3
State	18	148	360	11 000
Municipality	70	435	200	6 300
Private company	320	1047	320	13 300

In 260 measuring places radon the concentration was more than 400 Bq/m^3, and in 94 places it was more than 1000 Bq/m^3, the highest value being 13 300 Bq/m^3. The maximum radon concentration was found at a university, in sub-basement and it was not a regular workplace.14 % from state radon measurements, 12 % from municipalities and 18 % from private companies exceeded 400 Bq/m^3.

The classification of public buildings in Table 2, have in some cases a large range. Schools consist buildings from comprehensive schools to universities. The day-care centre is in general a large building, but there are few family dwellings included, too. STUK made a special study on radon concentrations in day-care centres, but these results are not included in this study. A day club for children belongs typically to parish activities in this study. Public room is a very heterogeneous group, from libraries to commercial places.

Table 2. *Statistics of radon measurements, public buildings*

Measuring place	Number of radon measurements	Average Bq/m^3	Maximum Bq/m^3	% > 400 Bq/m^3
School	228	400	13 300	20
Day-care centre	69	210	1 830	13
Day club for children	12	270	1 120	17
Public room	131	220	5 800	12

The customer has specified the measuring places. In many cases no classification is done and these measurements belong to the group Others.

Table 3. *Statistics of radon measurements, classified by type of measuring place*

Measuring place	Number of radon measurements	Average Bq/m^3	Maximum Bq/m^3	% > 400 Bq/m^3
Service or instrument room	132	710	11 160	37
Industrial premises	148	220	6 120	10
Office	518	220	5 240	12
Underground tunnel	124	660	11 000	32
Storage room	114	310	2 450	22
Others	570	260	13 300	14

There are plenty of measurements made at offices. This is the consequence of present measuring instructions. No regular workplaces are included in tunnels, but workers visit daily in some of the tunnels.

The amount of radon measurements in working places will increase slowly. It is possible to see from the postcodes and from the exceeding per cents that most of the radon measurements have been done in high radonrisk areas.

Results of short time radon measurements

Short time radon measurements have been made with Alpha Guard or, in some cases, with Pylon AB-5 measuring equipments. The measuring time is at least one week and the measuring period is one hour. STUK verified the radon concentrations during working time for 25 employers during the measurement seasons 2005-2007. Twenty six measuring places was the biggest amount for one employer, and this target consisted of two large buildings.

In fourteen cases employers had made radon measurements with alpha track method and received results that exceeded the action level 400 Bq/m^3. Short time measurements showed that in three cases the radon concentration was too high also during working time.

In eleven cases the radon concentrations were below the action level, in three places they sealed up chinks between the wall and the floor and adjusted timing of ventilation before measuring, and in nine places the existing ventilation took care of radon gas.

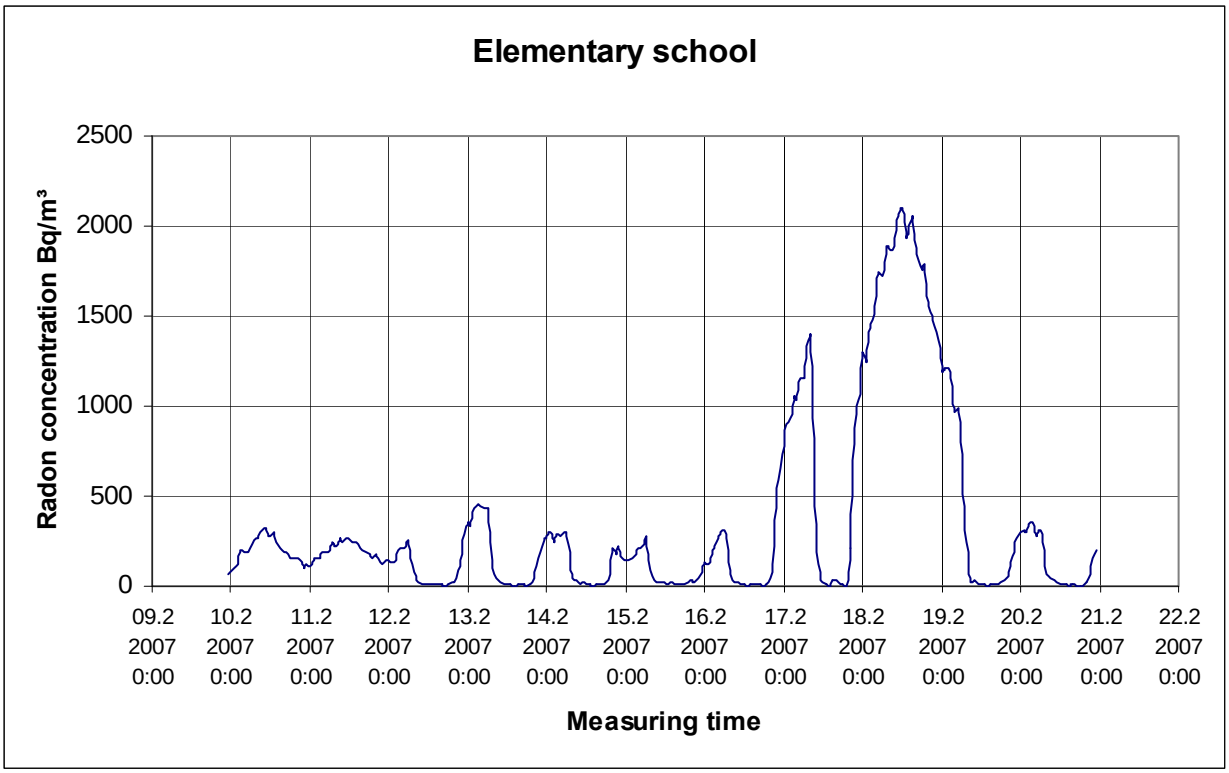
Short time radon measurements were done also in two underground cavities and five tunnels. There were regular working places in underground cavities, but in these cases the radon concentration was low. In tunnels the fitting of new equipments or renovations were started.

Radon measurements were made at four buildings because the owners were planning renovations, and they wanted to take radon into account beforehand.

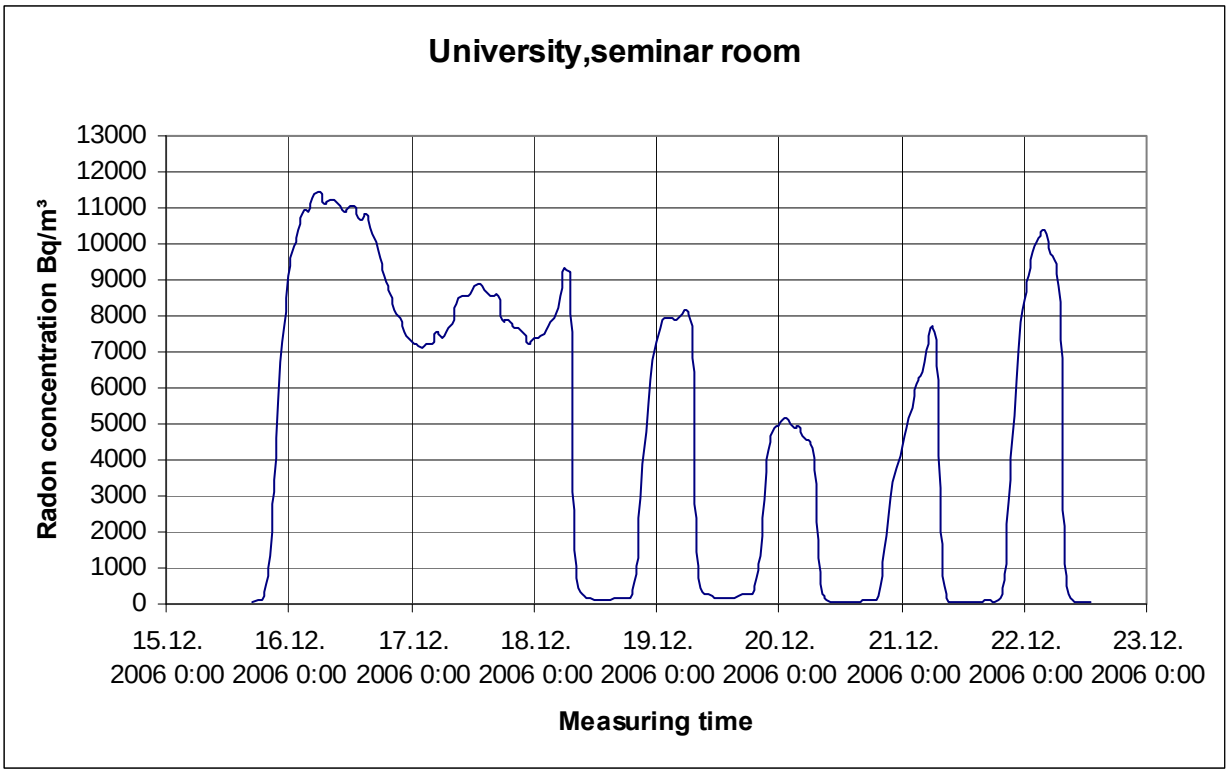
Examples of short time radon measurements

The first example is a new elementary school which is built in a gravel pit. They have three sub-slab suction systems, each for 500 m^2. They have also a good intake and exhaust ventilation with separate ventilation equipments. The first radon concentration measured with an alpha track method exceeded the action level 400 Bq/m^3. The radon concentrations with short time measurements were 300 Bq/m^3 for whole measuring time, ranging less than 20-2100 Bq/m^3, and less than 20 Bq/m^3 for the working time. There was an attempt to save energy, and the intake ventilation was closed every now and then. This created a high vacuum into the school building and a very high radon concentration at the same time.

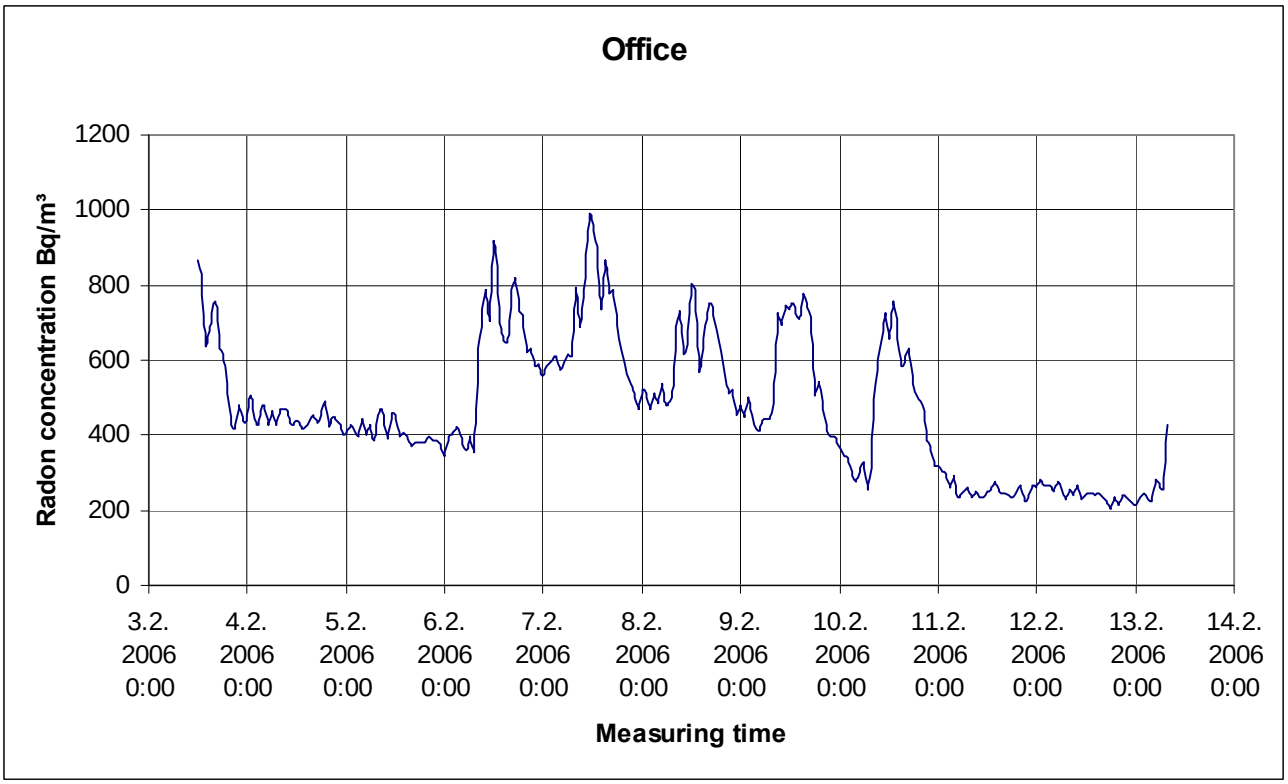
If the existing systems are used properly, one of the lowest radon concentrations among Finnish schools during working time is found there.



Our second example is a seminar room at a university. The building is located on granite rock and there is crushed granite aggregate under the floor. The radon concentrations with short time measurements were 4900 Bq/m^3 for whole measuring time, ranging 60-11 450 Bq/m^3, and 130 Bq/m^3 for the working time. This is the highest radon concentration that has been mitigated with existing ventilation.



The third example is an office, where the exhaust ventilation increases vacuum, and the daily radon concentration is higher than during night or week ends. They solved the problem with a new intake and exhaust ventilation equipment. The radon concentration is now about 10 Bq/m^3.



4 POSTERS

^{137}Cs uptake of forest berries

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Abstract: The variation of ^{137}Cs uptake by different forest berries was investigated by analysing berry samples collected during 1987–1989 in four sampling areas in northern Finland. The ^{137}Cs deposition after the Chernobyl accident in the study area varied from 1.6 to 7.3 kBq m^{-2}. The ^{137}Cs uptakes of blueberry (*Vaccinium myrtillus*) and lingonberry (*Vaccinium vitis-idaea*) were nearly similar, and that of crowberry (*Empetrum nigrum*) about 20 percent lower, while the ^{137}Cs uptake of cranberry (*Vaccinium oxycoccus*) was 3–4 times higher than that of the other berries. No clear annual variation was seen in the ^{137}Cs uptake of the berries picked in 1987, 1988 or 1989 at the same sites; the ^{137}Cs concentrations were at the same level or a slight increase was seen at some sites.

Introduction

Forest berries constitute an essential part of the Finnish diet. About 67% of adult people pick up forest berries in Finland. In good crop years people pick about 50 million kilos berries, of which 35 million kilos are used as personal home diet and 15 million kilos are picked commercially. Lingonberry is the most important Finnish forest berry, about 25 million kilos/year, followed by blueberry with about 15 million kilos/year (Moisio, 2006). The most productive forest type for berry picking is the heath forest. On average the greatest blueberry and lingonberry crops are found in the northern parts of Finland.

The sources of radiocaesium of forest berries in Finland are the atmospheric nuclear weapons testing in the 1950s and 1960s and the Chernobyl accident in 1986. The accumulated ^{137}Cs deposition in Finland has been estimated as 1.8 kBq m^{-2} by the end of 1985 (Saxén *et al*., 1987). The ^{137}Cs deposition after the Chernobyl accident in the study area varied between 1.6 and 7.3 kBq m^{-2} (Arvela *et al*., 1990).

The behaviour of radiocaesium in forests has been widely investigated. It is known that most decisive for radiocaesium uptake are the kind of humus layer, thickness and pH value, and soil properties. At different sampling sites the values of aggregated transfer factors can vary by two orders of magnitude for one plant species (Drissner *et al*, 1998). The aim of the present study was to investigate this variation in ^{137}Cs uptake by different species of forest berries and also to find out the annual changes during the years 1987–1989.

Material and methods

Sampling: The samples of the analysed forest berries were collected during 1987–1989 in northern Finland (Fig. 1). The study areas are situated in northern Finland, where more than 50% of forest berries marketed in Finland is picked. In sampling areas 1–3 (Pyhäjoki, Taivalkoski and Ylikiiminki) the Chernobyl fallout was 1.6–1.9 kBq m^{-2} (1.10.1986) and in the area 4 (Sotkamo) 7.3 kBq m^{-2} (Arvela *et al*., 1990). The studied forest berries were blueberry (*Vaccinium myrtillus*), lingonberry (*Vaccinium vitis-idaea*), cranberry (*Vaccinium oxycoccus*) and crowberry (*Empetrum nigrum*). The samples were provided by Department of Botany, University of Oulu, for a project studying crops of

forest berries in the areas of four municipalities (Table 1, Fig.1). The study areas are forests located in the Middle Northern Boreal vegetation zone. The sampling sites in the area represent dry, dryish and fresh forests and bogs. The forests in the study areas are mainly coniferous or conifer-dominated mixed forests. The soils of the pine dominant forests are mainly sand and those of the spruce-dominated forests till. The majority of these forest soils are to a greater or lesser extent acid and poor in nutrients.

Table 1. *Numbers of samples and sampling sites (given in paranthesis)*

Species	Pyhäjoki	Ylikiiminki	Taivalkoski	Sotkamo	Total
Blueberry	32 (4)	181 (6)	84 (6)	144 (6)	441 (22)
Lingonberry	183 (7)	380 (7)	208 (6)	247 (9)	1018 (29)
Cranberry	1 (1)	43 (8)	86 (7)	7 (1)	137 (17)
Crowberry	44 (6)	60 (7)	22 (2)	57 (3)	183 (18)

Figure 1. *Sampling areas. 1=Pyhäjoki, 2=Ylikiiminki, 3=Taivalkoski, 4=Sotkamo*

Measurements: The samples were frozen, stored in plastic bags, melted and dried in the oven (105 ˚C) until constant weight, and homogenised. The ^{137}Cs concentrations on dried homogenised samples were determined by gammaspectrometric measurement. The measurement errors were less than 10 percent, on average 5–6 percent.

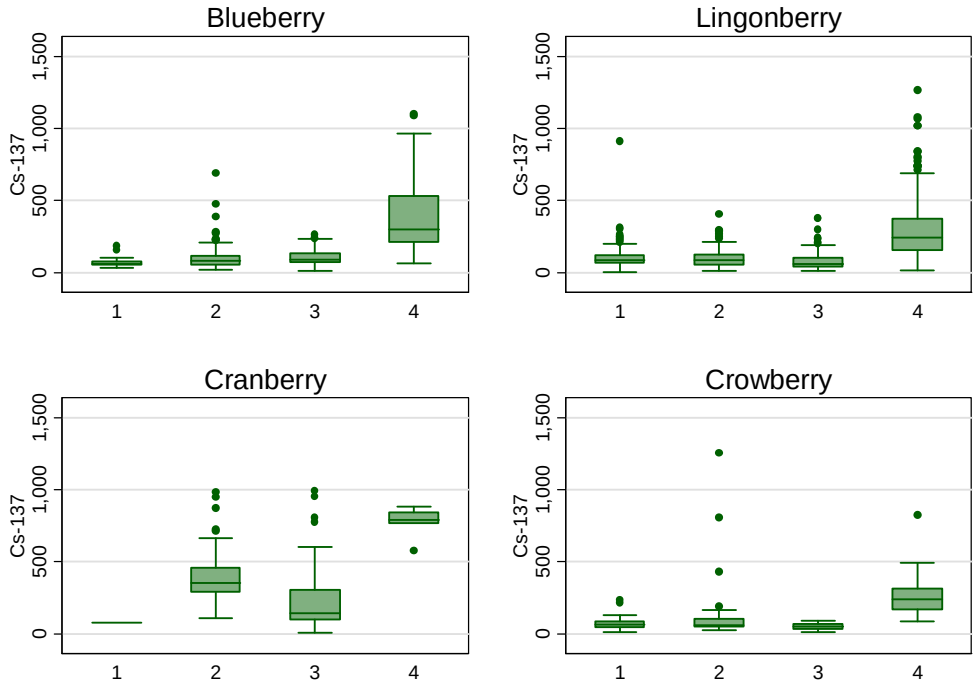
Data treatment: The mean, median and range of ^{137}Cs concentrations (Bq kg^{-1} dry weight) were calculated separately for the four municipalities and the whole area. The aggregated transfer factors (T_{ag}, Bq kg^{-1}/kBq m^{-2}) from deposition to berries were calculated for all the samples using mean regional deposition values of the study areas.

Results and discussion

^{137}Cs in berries. The ^{137}Cs concentrations in the berries are shown in Fig. 2. In Sotkamo (area 4), where the ^{137}Cs deposition was highest, the mean ^{137}Cs contents of all the berries were about fourfold compared to those in the other study areas. The great variation in the ^{137}Cs concentrations of the

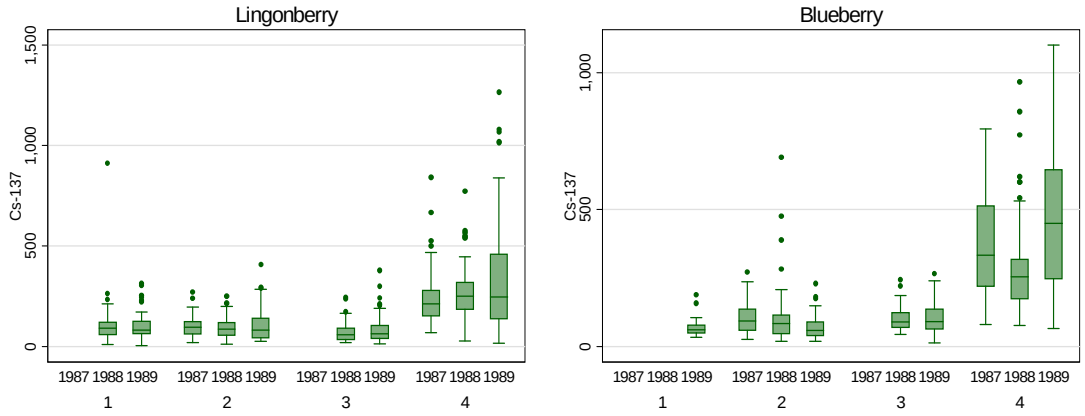
berries inside each area is in connection with the uneven deposition and the variation in the type of forest soil. No significant differences were seen in the ^{137}Cs concentrations of blueberries and lingonberries picked at the same sites inside the study areas, whereas ^{137}Cs levels in cranberries were 3–4 times higher and in crowberries about 20 percent lower than those in blueberries and lingonberries.

Figure 2. Median and range of ^{137}Cs concentrations (Bq kg^{-1} dry weight) of forest berries in the areas 1–4 (1=Pyhäjoki, 2=Ylikiiminki, 3=Taivalkoski, 4=Sotkamo) in 1987–1989.



The levels of the ^{137}Cs in the berries picked in 1987, 1988 or 1989 at the same sites showed no clear annual variation. The ^{137}Cs concentrations were at the same level or at some sites a slight increase was seen. The annual variation of the sampling areas 1–4, seen in Fig. 3, is probably due to the variation of the sampling sites inside the areas in successive years. The variation between the years depends also on the weather conditions in each particular year. The summer 1988 was an average summer, 1987 cold and rainy and 1989 warm and dry.

Figure 3. *Annual variation of ^{137}Cs contents (Bq kg^{-1} dry weight) of lingonberry and blueberry in the sampling areas 1–4 during 1987–1989.*

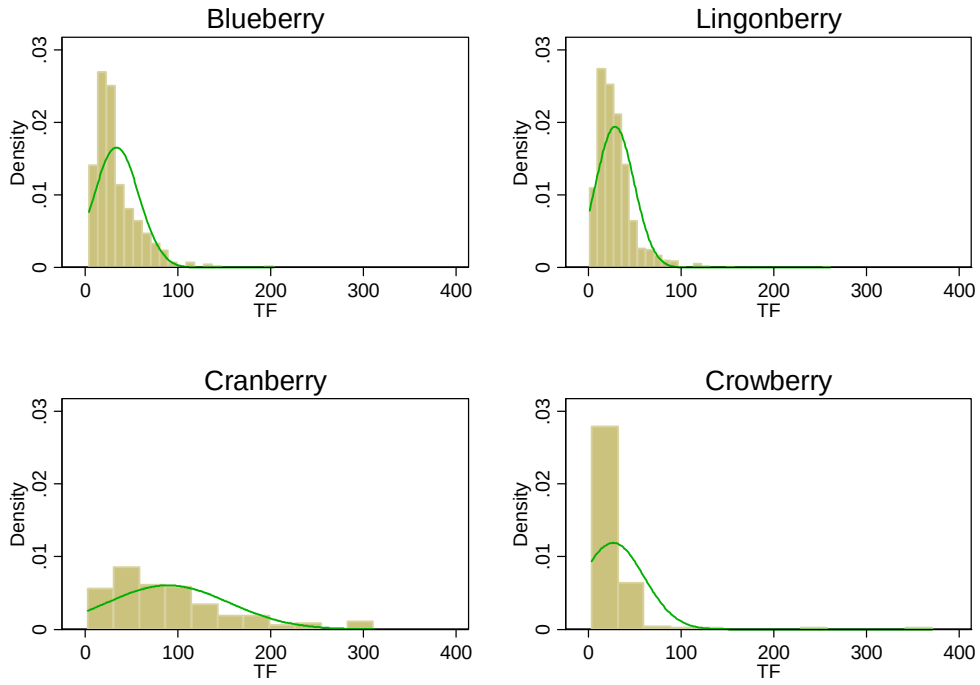


Transfer factors of berries. The aggregated transfer factors, i.e. the ^{137}Cs activity in dried berries (Bq kg^{-1}) divided by the ^{137}Cs activity of the total deposition (kBq m^{-2}), were calculated for all the samples by using mean regional deposition values of each study area. The distribution of the transfer factors for different species of berries is shown in Fig. 4. The average and median values for transfer factors (Table 2.) showed slight differences between the areas, which reflect the inaccuracy of the regional deposition values used and the variation of the growing conditions in the different sampling areas.

Table 2. *Aggregated transfer factors for berries in 1987–1989 (T_{ag} , Bq kg^{-1} dry weight / kBq m^{-2}). Median; mean; Q_1-Q_3 (Q_1=25th percentile, Q_3=75th percentile).*

Species	Pyhäjoki	Ylikiiminki	Taivalkoski	Sotkamo	All areas
Blueberry	18; 20;15-23	35; 45; 25-62	29; 34; 21-43	24; 28; 15-34	27; 34; 18-44
Lingonberry	25; 29; 18-35	28; 34; 17-44	19; 25;13-33	26; 28; 15-37	25; 29; 15-37
Cranberry	22	93; 92; 89-100	47; 74; 31-100	106; 122; 86-138	73; 89; 40-116
Crowberry	19; 21; 13-25	27; 29; 19-36	16; 15; 8-21	18; 33; 15-31	20; 27; 15-29

Figure 4. *Distribution of the aggregated transfer factors for forest berries (T_{ag}, Bq kg^{-1} dry weight / kBq m^{-2}).*



The deposition in forests is very unevenly distributed and variations of up to a factor of 10 may occur within a small area (Rűhm *et al*, 1996). One reason for the large variation in the transfer factors is that the berries for this study were picked inside the sampling areas also at sites not so typical to the species. The mean transfer factors for blueberry and lingonberry are similar to those reported earlier by Johanson *et al*, 1996, but a little lower for cranberry. The aggregated transfer factors of blueberry and lingonberry derived from the data of 1987–1989 are in the same range with those reported for the years 2000–2005 (Kostiainen, 2007), so no reduction in the ^{137}Cs uptake can be seen during the past twenty years.

Estimate of dose. The average consumption rate of forest berries in northern Finland is 13 kg per year (Markkula, 1997). In 1987–1989 the annual intake of radiocaesium from lingonberry and blueberry in northern Finland varied between 160–560 Bq, corresponding to a dose of 1.5–6 µSv.

Conclusions

The results of this study give the range of variation in ^{137}Cs uptake of forest berries in varying growing conditions during the first years after the deposition. The aggregated transfer coefficients given in this paper are well suited for a rapid estimate of contamination of forest berries in emergency situations.

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Changes in ^{137}Cs activity concentrations with time in various fish species and water in lakes of Finnish Lapland

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Abstract

The radioactive fallout in Finnish Lapland from the nuclear reactor accident in Chernobyl in 1986 was low, for ^{137}Cs in average 1000 Bq/m^2. The enrichment of radionuclides in the special arctic food chain like lake water - fish - man is however effective and the arctic and sub arctic environment is very vulnerable. The aim of this study was to find out how the situation of ^{137}Cs concentrations in Lakes of Lapland have changed after 1986 and what is the current situation. The ecological half-lives and the environmental factors witch influences the long-term behaviour of ^{137}Cs were estimated, correlations between fish and lake water analysed and concentrations factors calculated.

^{137}Cs concentrations were analysed in various fish species and corresponding lake waters from 1982 to 2007. Samples were taken from Lake Apukka, witch is situated near by the Artic Circle and from Lake Inari witch is the third largest lake in Finland located in the northern part of Lapland and important for freshwater fishing. The fish sampled consisted of pike, perch, burbot and roach. All samples were analysed by gamma spectrometry and the results of the measurements processed statistically.

Introduction

The radioactive fallout in Finnish Lapland from the nuclear reactor accident in Chernobyl in 1986 was low, for ^{137}Cs in average 1000 Bq/m^2 (Arvela et al., 1990). The enrichment of radionuclides in the special arctic food chain like lake water - fish - man is however effective and the arctic and sub arctic environment is very vulnerable. ^{137}Cs could still be measured after two decays from the accident in various fish species of the lakes. The ecological half lives, which seemed to be highly depended on lake type and the feeding habits of the fish species, could also been estimated in different lake types inside the same fallout area. Fish and water samples were collected from two different lakes, Lake Inari and Lake Apukka, which represent two different types of lakes from different drainage areas Paatsjoki and Kemijoki in Finnish Lapland. ^{137}Cs concentrations in fish and corresponding lake waters were analysed at the Radiation and Nuclear Safety Authority (STUK) in the Regional Laboratory in Northern Finland.

The aim of this study was to find out how the situations of ^{137}Cs activity concentrations in Lakes of Lapland have changed in time and what the current situation is. The ecological half-lives and the environmental factors witch influences the long-term behaviour of ^{137}Cs were also estimated.

Matherials and methods

Lake Inari, which is the third largest lake in Finland covering 1040 square kilometres, is situated in the North of Finnish Lapland (69 N, 28 E). Lake Inari is an oliotrophic arctic lake with rich fish fauna and it is important for freshwater fishing. Species of the samples were pike *(Esox lucius)*, perch *(Perca*

fluviatilis), trout *(Salmo trutta)*, whitefish *(Coregonous lavaretus)* and vendace *(Coregonus albula)*. The Lake Apukka is very different from Lake Inari. The lake is small, with an area 0,48 square kilometres and it is low and highly eutrophic. The fish fauna consist of pike, perch, roach and burbot *(Lota lota)*. Lake Apukka is situated in the Northern Finland near by the Arctic Circle. The locations of the lakes are shown in Fig.1.

Fish samples were caught regularly in Lake Inari 1963 - 1970, 1986 - 2007 and in Lake Apukka 1982 - 2007. Water samples were taken from year 1988 until 2005. ^{137}Cs was measured by gamma spectrometry.

Figure 1. *Locations of the lakes.*

Results and disussion

Fish

The results of ^{137}Cs in fish samples are shown in Fig. 2. Every single point represents the mean value of ^{137}Cs concentrations in studied specie per year. After the Chernobyl accident (CPP) in 1986 the highest ^{137}Cs concentrations were measured in pike, which is the top predator (piscivorous) in lake ecosystems and in species, which are both predators and non-predators (omnivorous) such as perch and trout. The lowest ^{137}Cs concentrations were found in whitefish and roach, which are non-predators (non-piscivorous).

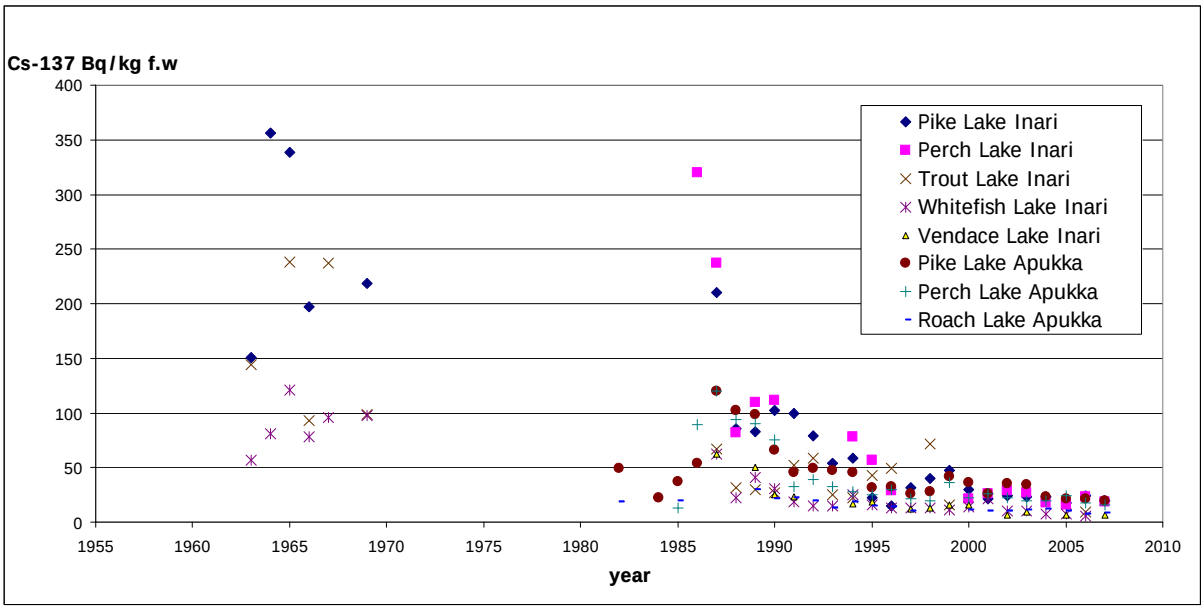

Figure 2. *^{137}Cs concentrations in various fish species in Lake Inari and Lake Apukka 1963 - 2007.*

Much higher ^{137}Cs concentrations were found in the beginning of 1960 due to the nuclear weapon tests. The number of the fish samples was limited before the CPP, but it can clearly been seen that ^{137}Cs concentrations were much higher compared to samples after the CPP accident. The variation of ^{137}Cs concentrations in fish, especially in the first years after the accident was large as seen in Fig. 3. Three years after the CPP accident (1987,-88,-89) the mean values of ^{137}Cs concentrations in fish were between 116- 237 Bq/kg f.w in Lake Inari.

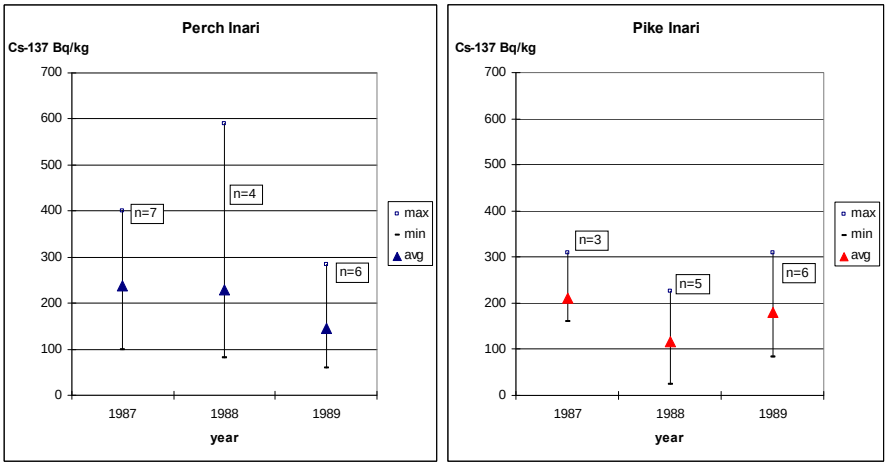

Figure 3. *The variation of ^{137}Cs concentrations in perch and pike in Lake Inari a few years after the CPP accident.*

Ecological half-lives in fish meat after the Chernobyl accident

Estimated ecological half-lives were determined by fitting a straight line to logarithmic scale scatter plot in each species. The best fitting line were built up by taking account the R^2 -value. The value should be as close as possible numerical value 1. Studied periods differenced so that in Lake Inari the 1. period was from 1987 to 1996 and the 2. period from 1997 to 2007 and in Lake Apukka the 1.

period was from 1987 to 1994 and 2. period from 1995 to 2007. R^2 -values varied in 1. period between 0,59 - 0,91 and in the second period between 0,23 - 0,62 in Lake Inari. R^2 -values were closely numerical value 1 in Lake Apukka where values varied between 0,59 - 0,91 in the 1. period. Second period were not linear as it was in Lake Inari.

Estimated ecological half-lives were calculated in three different fish species in both lakes (Table 1). Pike and perch were studied both in Lake Inari and Lake Apukka, but whitefish only in Lake Inari and roach only in Lake Apukka. Whitefish and roach were classified as a non-piscivorous species. Ecological half-lives varied between 2,2 - 4,5 years in 1. period in all species in both lakes. In the second period the ecological half-lives differed much between lakes. Ecological half-lives were near the physical half-lives (30 year) in piscivorous and omnivorous species in Lake Apukka.

Table 1. *Estimated ecological half-lives [a] and R squared value (in parenthesis) in Lake Inari and Lake Apukka.*

Lake/Specie/ Time Period	Inari			Apukka		
	Pike	Perch	Whitefish	Pike	Perch	Roach
1. period	**3,0** (0,79)	**4,5** (0,78)	**2,2** (0,60)	**4,6** (0,86)	**3,2** (0,91)	**4,3** (0,59)
2. period	**9,9** (0,62)	**19,9** (0,23)	**6,7** (0,54)	**24,6 (**0,35)	**28,2** (0,26)	**20,9** (0,47)

Freshwater

Freshwater data were available from year 1988 to 2005 in both lakes. Variation in ^{137}Cs concentrations was large a few years after the CPP accident. In Lake Inari the minim and maxim values were 0,3 and 22 Bq/m^3 and in Lake Apukka 0,7 and 12 Bq/m^3. ^{137}Cs levels decreased from mean value 10 (1988-1993) to 1 Bq/m^3 in Lake Inari and from mean value 5 to 1 Bq/m^3.

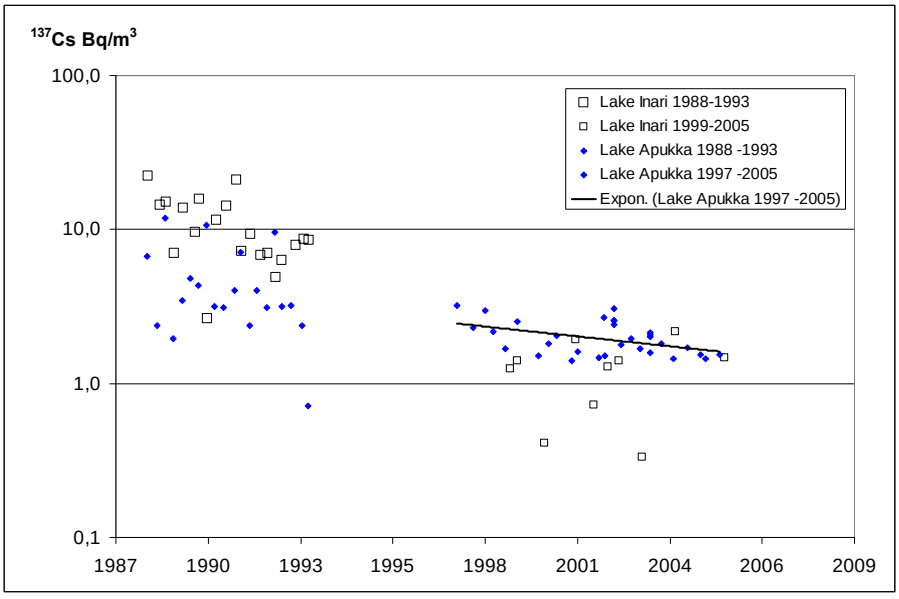

Figure 4. *Freshwater data from Lake Inari and Apukka*

Conclusions

^{137}Cs concentrations were on early studies (Saxen and Sundell, 2005) increased to a maximum in the second, third or even fourth year after the CPP accident. In Northern Finland, where the fallout was much lower, there was not seen this effect.

Differences in ecological half-lives in fish were not much depending on the type of the lake or fish specie during the first years after the accident. Large differences were seen 8 years after the accident. In the lake which is olotrophic, like Lake Inari, the ecological half lives were in the second period much lower than in the lake which is eutrophic, like Lake Apukka, especially in the species like pike and perch. ^{137}Cs concentrations were also decreased more rapidly in Lake Inari than Lake Apukka according the freshwater data. The main reason to long half-lives in fish in Lake Apukka is a characteristic of the lake vicinity that keeps levels up year after year.

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Innhold av radiocesium i jord og sopp i Øst-Finnmark

Hanne Solheim Hansen, Inger M. H. Eikelmann, Bredo Møller

Statens Strålevern

Abstract

I Øst-Finnmark stammer det meste av radiocesium fra bombenedfallet fordi det kom lite nedfall i forbindelse med Tsjernobyl-ulykken. Innholdet av radiocesium i ulike nivå av økosystemet målt over tid kan derfor brukes til å predikere hvordan nedfallet fra Tsjernobyl vil oppføre seg i et langtidsperspektiv.

I Sør-Varanger kommune har vi 5 lokaliteter der vi med noen års mellomrom har tatt prøver av jord og sopp siden 1993. På hver lokalitet ble det tatt 5 jordprøver. For å kjenne areal og volum av jordprøvene ble det brukt en sylinder med kjent diameter og høyde. Soppen ble plukket innenfor det området som jordprøvene ble tatt. Når det var tilstrekkelig mengder av enkeltarter av sopp ble det tatt rene artsprøver, ellers ble det tatt samleprøver av kremler (*Russula spp.)*, risker (*lactarius spp.)*, slørsopp (*Cortinarius spp)* og rørsopp (*Boletus spp og Leccinum spp.).* Det er mulig å få nok materiale av rødbelteslørsopp (*Cortinarius armillatus)*, siden den er giftig og heller ikke blir spist av beitedyr. Ale prøver ble tørket og malt før prøvene ble målt for innhold av Cs-137 på Ge-detektor.

Innholdet av radiocesium i jord var 30-50 Bq/kg d.w. Innhold av radiocesium i sopp varierte mellom artene. Det var generelt høyt innhold i slørsoppene, mens det var lavere innhold i kremler, risker og rørsopp. I rødbelteslørsopp og rimsopp var nivået på mellom 3000 og 4000 Bq/kg d.w., mens det var mellom 300-1000 Bq/kg d.w. for de andre artene. Nivået av radiocesium i soppene ser ut til å ha holdt seg ganske stabilt de 14 siste årene.

Effectiveness of Sr-binders tested using an *In Vitro* model

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Abstract

The radioisotope ^{90}Sr is a fission product that in biological systems will behave like calcium and accumulate in bone. To obtain food products, particularly milk, with low radiostrontium levels, the use of Sr-binders in animals may be required. We have tested some zeolites that may be used as Sr-binders in animal production.

Preliminary studies demonstrated that only zeolite A(Na) and zeolite P would be of practical interest. Given at a rate of up to 30 g/d to goats the Sr transfer to milk was reduced by 40%. However, because of the chemical properties of the zeolites, they may also bind other cat ions like Ca and Mg, both important minerals for animals. We have therefore tested the binding kinetics of Sr, Ca and Mg by the zeolites in rumen liquid from cows.

The zeolite A(Na), at 0.5 % in rumen liquid, pH 7.3, bound 98 % of the Sr. When pH was reduced to 2.5, similar to the pH in the abomasum, no Sr was bound to the zeolites. This was also the case at pH 3 and 4. At pH 5 ca 10 % of the Sr was bound. Theses levels of pH simulated the conditions in the small intestine. To simulate the conditions in the large intestine the pH was increased to pH 8, and subsequently the Sr-binding was elevated to 80 %. The zeolite P demonstrated similar binding kinetics as the zeolite A(Na). The zeolite A(Na) bound ca 90% of the Ca, while zeolite P bound ca 65 % of the Ca in the rumen liquid mixture. For all treatments and binders the binding of Mg was less than 20 %.

We conclude that zeolite A(Na) has promising properties as a Sr binding agent in animals, however, because the zeolite A(Na) demonstrated high affinity also to Ca, further studies of the binding of other minerals are required before zeolite A(Na) should be taken into practical use.

5 Thorium lecture

Thorium som energiressurs. Fordeler og ulemper

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Interessen for thorium som brensel i reaktorer har variert gjennom de siste 50 år. På 50- og 60-tallet så man for seg en snarlig mangel på uran, og det ble drevet en intens forskning med henblikk på å utnytte verdens thoriumressurser. Også Norge deltok aktivt i dette arbeidet gjennom sin deltagelse i Dragon-prosjektet (1966 - 1973) i Storbritannia. Flere titalls reaktorer av vidt forskjellige konstruksjon var i drift med thoriumbrensel fra midt på 60-tallet til ut i 80-årene. Flesteparten var små forsknings-reaktorer, men det ble også bygget relativt store reaktorer for kraftproduksjon. De mest kjente er Fort St. Vrain i USA og THTR i Tyskland, som produserte henholdsvis 330 og 300 MW elektrisitet. Store funn av uran utover i 70-årene gjorde at interessen for thorium avtok, og for tiden er det kun India som satser for fullt på kjernekraft fra thorium.

I jordskorpa finnes det 3 - 4 ganger så mye thorium som uran, og i tillegg er det mulig å utnytte bortimot 100 % av det i termiske reaktorer, mens det tilsvarende tallet for uran bare er omkring 1 %. (I de mer avanserte hurtigreaktorene er det mulig å benytte seg av nær 100 % av uranet også.)

Thoriuminteressen i Norge blusset for alvor opp for et par år siden da professor Egil Lillestøl foreslo at Norge skulle bygge en kraftreaktor basert på akseleratorteknikk, den såkalte Rubbiareaktoren. Denne benytter en underkritisk reaktor, det vil si en reaktor som har for lite brensel til å kunne holde kjedereaksjonen i gang på egen hånd. Ekstra nøytroner for å drive prosessen tilføres fra en kraftig protonstråle som bombarderer bly og river løs nøytroner fra blykjernene.

Norge har store forekomster av thorium, hovedsakelig lokalisert i Fensfeltet ved Ulefoss. Foreløpige analyser konkluderer med at thoriumet foreligger i en form som gjør det vanskelig å utvinne med dagens teknologi.

Thorium som drivstoff har både fordeler og ulemper sammenlignet med uran. Brenselet er mer stabilt, har bedre varmeledningsevne og mindre varmeutvidelse. Dessuten produseres det svært lite av de såkalte transuraner, som er langlivede avfallsprodukter. Det fører til at avfallet trenger kortere lagringstid, i underkant av 1000 år, sammenlignet med flere titusen år for uprosessert uranbrensel. (Her må det tilføyes at den korte lagringstiden forutsetter at brenselet reprosesseres og uran og thorium føres tilbake til reaktoren.) En akseleratordrevet reaktor har dessuten unike egenskaper med hensyn til "brenning" av radioaktivt avfall ved hjelp av fisjon og transmutasjon.

En ulempe ved thorium er at det er mer komplisert å produsere brenselet, og det er heller ikke enkelt å håndtere det brukte brenselet, både på grunn av dets kjemiske stabilitet og et høyt strålenivå. På den annen side kan dette også sees på som en fordel, siden det vanskeliggjør anvendelse til våpenformål.

6 Emergency Preparedness session

Fortbildning för sjukhusfysiker inom nationell strålskyddsberedskap i Sverige

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Abstract

Vid en allvarlig händelse, t.ex. olycka, brand etc., är det antingen räddningstjänst, polis eller ambulans som kommer först till platsen. Om man kan misstänka att det finns radioaktiva ämnen på olycksplatsen, t.ex. olycka med transport av radioaktiva ämnen kan räddningstjänsten i Sverige idag eventuellt utföra enkla dosratsmätningar. Om det inte finns indikation på att det skulle finnas radioaktiva ämnen på olycksplatsen är det osannolikt att någon mätning görs. Finns det en konstaterad eller misstänkt förekomst av radioaktiva ämnen på olycksplatsen ska räddningstjänsten kontakta den nationella strålskyddsmyndigheten (f.d. SSI), som ska kunna ge expertstöd i någon form, antingen via ett bakre stöd eller med utsänd expertis på skadeplatsen. Det är i sådana sammanhang också troligt att räddningstjänsten via krisledningscentralen (SOS-alarm) tar kontakt med en sjukhusfysiker som får agera strålskyddsexpert på plats.

Sjukhusfysikern har en nyckelroll när det gäller att hantera problemen med skadade och kontaminerade personer vid akutmottagningar och efterföljande sjukvård vid en radiologisk nödsituation. För att ge sjukhusfysikerna i Sverige möjlighet till kompetensutveckling och vidareutbildning inom strålskyddsberedskap utanför sjukhusens normala verksamhet ges årligen sedan 2006 en s.k. CPD-kurs. Denna behandlar beredskapens organisation, samverkan mellan myndigheter, åtgärdsstrategier samt strålskyddsexpertens uppgifter i fält. En påbyggnadskurs inom detektorer och mätteknik kommer att ges från och med 2008. Utbildningen ska intensifieras under 2009-2011 genom att successivt införa nya kurser och den svenska strålskyddsmyndighetens målsättning är att huvuddelen av alla sjukhusfysiker har genomgått utbildning i krisberedskap 2011. Flera av kurserna är, i mån av plats, öppna för deltagare från övriga nordiska länder.

De svenska sjukhusens beredskap för olyckor med strålning och kontaminerade personen uppvisar en mycket stor variation. På vissa platser finns bra beredskap med fastställda larmvägar till sjukhusfysiker; på andra platser är strålningsolyckor inte medtagna alls i den lokala katastrofmedicinska planen. Vid en olycka med radioaktiva ämnen kan det alltså komma att ställas krav på att en sjukhusfysiker, som inte fått möjlighet till övning och som inte har tillgång till lämpliga mätinstrument, ska kunna ge expertstöd och/eller utgöra mätpatrull vid en olycksplats eller i samband med sanering av kontaminerade patienter.

Nationell strålskyddsberedskap i Sverige

Händelsen den 28 mars 1979 vid kärnkraftverket Three Mile Island i USA fick direkta konsekvenser för den svenska beredskapen mot kärnenergiolyckor. Trots att mycket små mängder radioaktiva ämnen läckte ut till omgivningen visade olyckan att även det som av dåtidens expertis bedömdes så osannolikt faktiskt kunde hända. En utredning vid SSI om effektivare beredskap ledde till att beredskapen förstärktes såväl i kärnkraftlänen som vid SSI. Dessutom tog regeringen beslut om att alla svenska kärnkraftverk skulle förses med utsläppsbegränsande filter. Nästa steg i utbyggnaden av beredskapen kom som en direkt följd av Tjernobylolyckan den 26 april 1986 och innebar att samtliga länsstyrelsers roll stärktes. Länsstyrelserna i kärnkraftlänen har nu en omfattande beredskapsorganisation som övas regelbundet. En SSI-utredning efter Tjernobylolyckan ledde också bland annat till att SSI:s gammastationer utökades från 25 stycken till 37 samt automatiserades och nu larmar vid förhöjda strålningsnivåer. Gammastationerna hade från början initierats av Rolf Sievert för att registrera det radioaktiva nedfallet från de atmosfäriska kärnvapenproven och avlästes då manuellt. I samband med en planerad modernisering har antalet nu minskats till 32 stycken. Andra åtgärder var att nationella mätresurser samordnades och att intensimetrar för återkommande dosratsbestämningar på särskilda lokala mätplatser delades ut till kommunerna.

Strålskyddsberedskapen är nu på väg att byggas ut ytterligare för att även kunna hantera händelser utanför kärnkraftsindustrin. Eftersom flertalet av dessa händelser inte inträffat är det inte så lätt att veta vad man ska ha beredskap mot och vi har alltså att göra med en risk där såväl sannolikheten som konsekvensen är okänd. Bästa sättet att skaffa sig förmåga att hantera även dessa händelser är att ständigt ha en god mätberedskap samt kompetens att kunna utföra och korrekt tolka mätningar och därifrån ge råd om fatta beslut om motåtgärder. Trots att risken i många fall är okänd kan man ändå identifiera följande tänkbara scenarier som har kommit att ligga till grund för planeringen av den svenska nationella strålskyddsberedskapen:

- reaktorolycka
- kärnvapennedfall i Sverige – från kärnladdningsexplosion utanför landet
- terrorism – exempelvis smutsig bomb, avsiktlig kontaminering av vatten eller livsmedel, strålkälla som placeras ut
- transportolyckor
- industriella olyckor – exempelvis brand; omfattar även sjukhus och universitet
- satelliter – som får sin elförsörjning via kärnreaktor eller radioaktiv strålkälla

Om man frågar sig hur samhället klarar en antagonistisk attack med strålkällor i någon form och hur sannolikt det är att en sådan händelse skulle inträffa kan man först konstatera att ”vardagsolyckor” med radioaktiva ämnen nästan aldrig inträffar. De funktioner i samhället som ska hantera olyckor, dvs. räddningstjänst, polis och amulanssjukvård saknar därför praktisk erfarenhet. Detsamma gäller för kommunerna, och särskilt de kommuner som inte är berörda av kärnkraftsberedskapen övar aldrig händelser med strålning. För det andra finns det endast ett fåtal personer i landet som har praktisk erfarenhet av att fungera som strålskyddsexperter i händelse av strålningsolycka. Till dessa hör vissa sjukhusfysiker som skaffat sig praktisk erfarenhet i sin yrkesutövning, men grundutbildningen för sjukhusfysiker innehåller för närvarande mycket litet om katastrofstrålskydd.

För att klara svåra påfrestningar på samhället i fredstid samverkar myndigheterna i sex samverkansområden och inom ett av dessa har SSI identifierat fyra långsiktiga mål för tiden fram till

och med 2008: *kompetensförsörjning*, *ledningsförmåga*, *mätning och analys*, samt *stöd till först-på-plats* (”first responders”). Det första målet är att kvalificerade experter finns att tillgå vid en nukleär eller radiologisk händelse (RN-händelse). För att närma sig detta mål görs nu en satsning på utbildning genom att Lunds och Göteborgs universitet tilldelats medel av SSI för att inrätta lektorstjänster inom ämnet katastrofstrålskydd. Innehavarna av dessa tjänster förväntas bedriva forskning, handleda doktorander samt utveckla kurser inom strålskyddsberedskap. I samband med detta görs även en satsning på beredskapsrelaterade doktorandtjänster och finansiering av CPD-kurser (Continuous Professional Development). Detta arbete kommer att beskrivas mera utförligt senare.

Ledningsförmågan, dvs. den nationella strålskyddsberedskapens förmåga att leda, samordna, samverka och informera, har stärkts t.ex. genom att två nya beredskapscentraler byggts upp. Utbyggnaden av mät- och analysresurserna har skett genom att SSI skrivit avtal med olika aktörer att fungera som s.k. beredskapslaboratorier. Dessa avtal gäller dock den nationella strålskyddsberedskapen och inte den beredskap som t.ex. kommuner och länsstyrelser i kärnkraftlänen upprätthåller eller den beredskap som finns hos räddningstjänst, polis och sjukvård. De flesta av beredskapslaboratorierna kan utföra fältmätningar (med t.ex. terränggående fältlaboratorier) samt laboratoriemätningar. Målsättningen att kunna ge adekvat stöd till ”first responders” innebär att utbilda personal samt att etablera ett expertstöd som ska finnas till hjälp vid en händelse.

Vad har sjukhusfysikern för roll?

Enligt socialstyrelsens kompetensbeskrivningar för sjukhusfysiker (Socialstyrelsen, 2001) ska sjukhusfysikern bland annat kunna göra insatser vid strålningsolyckor och katastrofer. Detta sammanfattas i följande punkter:

> *En sjukhusfysiker behöver med utgångspunkt i de funktioner som anges för yrkesområdet* medverka i samhällets beredskap mot strålningsolyckor *och därvid kunna*
>
> - *delta som rådgivande expert i samhällets strålskyddsberedskap vid olyckor och katastrofer (t.ex. strålning i samband med transporter, kärnenergiolyckor m.m.)*
> - *delta i sjukvårdens beredskap för katastrofer vid t.ex. mätning och dekontaminering av inkommande patienter från en strålningsolycka, dosuppskattningar och riskbedömningar*
> - *fungera som strålnings- och mätkunnig expert och rådgivare vid mindre incidenter och olyckor med strålning och radioaktiva ämnen.*

Beredskapslaboratorierna ska upprätthålla beredskap att kunna agera enligt fastställd uppgiftsförteckning och har en s.k. uppstartstid på 1-3 dygn, beroende på uppgift. Med uppstartstid avses den tid det får ta från det att personalen kommer till beredskapslaboratoriet till det att uppgiften är färdig att utföras. Av resursskäl anges emellertid inte hur lång tid det får ta innan beredskapslaboratoriets kontaktperson svarar på ett larm från SSI eller hur länge det får dröja innan personal är på plats. Beredskapslaboratoriet kan alltså inte ersätta andra viktiga aktörer beträffande strålskyddsexpertis i initialskedet efter en RN-händelse.

Vid en händelse som omfattar spridning, eller risk för spridning, av radioaktivt material till omgivningen är det antingen räddningstjänst, polis eller ambulans som kommer först till platsen. Om det finns indikationer på att det finns radioaktiva ämnen på olycksplatsen, t.ex. transportolycka med

skyltad bil, kan räddningstjänsten göra en första indikeringsmätning med SRV2000 (vissa räddningsfordon är utrustade med instrument). Om ingen förvarning eller annan särskild indikation som anges i Svenska räddningsverkets Åtgärdskalender (Socialstyrelsen m.fl., 2006) finns om förekomst av radioaktiva ämnen, är det mycket osannolikt att någon av aktörerna som är först-på-plats skulle utföra en dosratsmätning. För att få expertstöd på plats ska, enligt planerna, TIB (Tjänsteman I Beredskap) vid SSI kontaktas för att kunna ge telefonsupport men det är också troligt att man via krisledningscentralen (eller sjukhuset) tar kontakt med en sjukhusfysiker. Det skulle alltså kunna inträffa att någon sjukhusfysiker får agera strålskyddsexpert på skadeplatsen, även om det i första hand förväntas att sjukhusfysikern får utföra kontaminationskontroll vid akutmottagningen.

Vad finns det då för beredskap vid krisledningscentralerna och sjukhusen för att hantera händelser med inslag av radioaktiva ämnen och personkontamination och hur går larmvägarna till sjukhusfysikern? Svaret på denna fråga kommer att bli mycket varierande beroende på vilken krisledningscentral och vilket sjukhus man avser. I Sverige är det en stor variation mellan landsting/regioner och även mellan enskilda sjukhus. På vissa platser finns fastställda larmvägar till sjukhusfysiker, där akutmottagningen har aktuella telefonlistor och mätinstrument; på andra platser nämns inte ens strålningsolyckor i den lokala katastrofmedicinska planen. Om något händer kan det alltså ställas väldigt stora krav på att en sjukhusfysiker som inte övat, inte har tillgång till adekvata mätinstrument och som kanske kallas med väldigt kort varsel, ska kunna fungera som expertstöd och/eller mätpatrull vid en olycksplats eller i samband med sanering av kontaminerade patienter.

CPD - kurserna

De svenska sjukhusfysikerna har nu möjlighet att gå flera olika CPD-kurser som planeras i samråd med kursrådet vid Svenska sjukhusfysikerförbundet. Under 2008 kommer tre kurser att ges och planen är att intensifiera utbildningen under 2009-2011. SSI:s målsättning är att huvuddelen av alla sjukhusfysiker år 2011 har genomgått utbildning i krisberedskap. Den första kursen i det kursprogram som kallas *Förbättrad nationell beredskap mot radiologiska och nukleära nödsituationer* gavs under hösten 2006, i samband med den stora nationella mätövningen DEMOEX, och följs under 2008 av två nya kurser. Dessa kurser presenteras nedan.

Kurserna finansieras med krisberedskapsmedel via SSI och är avgiftsfria för sjukhusfysiker som är verksamma i Sverige. Resa till och från kursen, samt eventuell lön under kurstiden, bekostas av kursdeltagarna eller deras arbetsgivare men SSI bekostar kost och logi samt annan kommendantur. Deltagare från övriga nordiska länder är välkomna men får då bekosta kost och logi själva.

Krisberedskap och strålskydd i radiologiska och nukleära nödsituationer

Denna kurs ges under tre dagar och utgör en grundkurs i strålskyddsberedskap. Kursen har nu getts 2006 och 2007; en ny kursomgång planeras till hösten 2008. Under kursen behandlas hur den svenska krisberedskapen är organiserad, vilka hotbilder som finns, åtgärder vid olika typer av händelser samt vilka roller olika aktörer har inom beredskapen. Dessutom ingår en översiktlig genomgång av olika mättekniker som är av betydelse inom strålskyddsberedskapen. Utbildningsmålen är att deltagarna ska få

- kännedom om vilka hotbilder som finns när det gäller bestrålning och spridning av radioaktiva ämnen
- kunskap om åtgärder vid bestrålning och spridning av radioaktiva ämnen, samt sanering

- kännedom om hur den nationella krisberedskapen med avseende på strålningsolyckor är organiserad
- kunskap om risker och riskkommunikation
- kännedom om vilka aktörer som finns inom krisberedskapen och hur dessa samverkar
- färdigheter i att använda de handhållna mätinstrument som utnyttjas i organisationen
- kännedom om persondosimetri inom krisberedskapen

Som förberedelse till kursen får deltagarna till uppgift att förbereda en presentation som behandlar en inträffad olycka av radiologisk eller nukleär karaktär. Detta ligger sedan till grund för en diskussion om hotbilder under första dagen. Examinationsuppgiften består i att varje kursdeltagare reflekterar över hur krisberedskapen inför strålningsolyckor utanför sjukvården ser ut inom den egna verksamheten. De ska då diskutera hur den egna verksamheten bör utformas avseende mottagning av skadade, dimensionering, tillgängliga instrument, genomförande av mätning och personsanering, skydd av lokaler, samverkan etc. Detta sammanfattas i en skriftlig rapport.

Kursvärderingarna har visat att det finns en efterfrågan på mer praktiska kunskaper kring radiometrin och mätmetoder inom strålskydd och detta har vi tagit hänsyn till när vi utvecklat ytterligare kurser inom ämnesområdet.

Detektorer och mätmetoder inom strålskydd och beredska*p*

Kunskap om detektorers karakteristik och erfarenhet av hur de vanligaste indikeringsinstrumenten inom strålskydd och beredskap fungerar, är en förutsättning för att medicinska strålningsfysiker ska kunna agera säkert i sin yrkesroll i händelse av en radiologisk eller nukleär nödsituation. Denna kurs består av två delar, där den första delen behandlar teorin bakom olika strålningsdetektorers karakteristik och en kortfattad beskrivning hur dessa material idag kombineras med modern elektronik för att få ut optimal prestanda. Den andra delen utgörs av ett antal praktiska laborativa moment under två dagar, dels i en strålningsmiljö som representerar en kärnteknisk anläggning, dels i en fältmässig miljö för att lokalisera, identifiera och kvantifiera strålkällor. Indikering av olika exponeringssituationer och strålningsmiljöer kräver varierande typer av detektorer, och i denna kurs ska en orientering göras över instrumentens olika tillämpbarhet i viktiga scenarier. Kursen gavs första gången under våren 2008 i samarbete med Barsebäcksverket och Medicinsk strålningsfysik i Lund (Lunds universitet).

Kursens mål är att deltagarna ska

- få en orientering om mätomfång, och grundläggande signalrespons hos de vanligaste typerna av indikeringsinstrument
- få en orientering om kvalitetsmått på detektorprestanda
- kunskap om hantering av mätosäkerheter och vilka faktorer som påverkar signal-till-bakgrundsvariation för t.ex. gammaspektrometri
- få basal handhavandeförmåga för ett antal viktiga strålskyddsinstrument
- få en orientering om mät- och analysmetoder av radiometriska data

Förberedelserna inför kursen består av att deltagarna ska inventera vilka strålskyddsinstrument och stationära detektorsystem som kan användas i en radiologisk och nukleär nödsituation, samt att undersöka vilken kvalitetskontroll som förekommer för de instrument som kan användas för indikering. Kursen examineras genom att deltagarna utför en enskild uppgift som lämnas in skriftligt.

Strålskydd vid katastrofmedicinska insatser

Sverige skall enligt de särskilda övergripande målen för beredskapen avseende nukleära och radiologiska nödsituationer ha en nationellt och internationellt väl samordnad beredskap för att förebygga, identifiera och möta nukleära och radiologiska hot. Strålskyddsberedskapen skall vid sådana händelser arbeta för att i första hand förhindra akuta skador på människor genom att hålla stråldoser under relevanta tröskelvärden. Detta förutsätter en väl fungerande beredskapsorganisation vid landets sjukvårdsinrättningar som i RN-händelser ska tillhandahålla ett professionellt medicinskt omhändertagande av externt och/eller internt exponerade människor, identifiering av strålkällor, bedömning av stråldoser och risker och säkerställandet av strålskyddet för sjukvårdspersonalen.

Denna beredskap kräver därför att det inom landets sjukvårdsorganisation finns ingående teoretiska och praktiska kunskaper om strålskydd inom katastrofmedicin . Kunskaper inom detta område hos berörd medicinsk personal (främst inom akutsjukvården) och sjukhusfysiker är avgörande för kunskapsspridningen om strålskyddsberedskap inom den lokala beredskapsorganisationen, och för uppbyggnaden av en fungerande beredskap. Nya hotbilder har uppmärksammats de senaste åren, t.ex. enskilda terrorattentat eller annan avsiktlig spridning av farliga ämnen, vilka ställer delvis nya krav på kunskap och praktisk erfarenhet och därmed skapar ett behov av vidareutbildning. I denna vidareutbildning bör ingå en kurs i medicinskt omhändertagande och strålskydd i samband med radiologiska och nukleära nödsituationer, med fokus på rollfördelningen mellan strålskyddsexpert samt vård- och läkarpersonal.

Kursen, som kommer att ges under hösten 2008, planeras i samarbete med Katastrofmedicinskt centrum vid Landstinget Östergötland/Linköpings universitet och Avdelningen för medicinsk radiofysik vid Linköpings universitet. Preliminärt kommer kursen att innehålla:

- beskrivning av sjukvårdsorganisationens beredskap mot CBRNE-händelser (internationellt, nationellt, regionalt och lokalt)
- olika aktörers uppgifter från skadeplats till akutmottagning och eventuell specialistvård
- riktlinjer för sanering och personavsökning
- strålskyddsdosimetri – fallstudier, persondosimetri samt retrospektiv dosimetri
- bio- samt interndosimetriska metoder
- medicinskt omhändertagande – demonstration och rollspel
- principer för sortering och prioritering vid sjukhus
- avsökning, sanering och provtagning på kontaminerade individer
- genomgång av expertstöd inom strålskydd till medicinsk personal vid samtidig behandling av trauma
- övning vid akutmottagning.

Framtidsplaner

Hittills har omkring 30 sjukhusfysiker deltagit i den första kursen ”Krisberedskap och strålskydd i radiologiska och nukleära nödsituationer” och omkring 20 har deltagit i ”Detektorer och mätmetoder inom strålskydd och beredskap”. Till detta kommer ca 15 doktorander som också läst dessa kurser. Kurserna utgör en god grund till att förbättra förmågan i Sverige när det gäller att hantera händelser som innebär risk för bestrålning med joniserande strålning. Till hösten 2008 beräknas ytterligare ett tiotal sjukhusfysiker kunna gå den planerade kursen i medicinsk omhändertagande i RN-händelser. Vår målsättning är att dessa kurser på sikt ska utökas till ett helt kurspaket inom strålskydd för samhällets behov, och som kan modifieras till att även ingå i universitetens kursutbud.

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NordRisk: Practical risk assessment for long-range atmospheric transport of radionuclides

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Abstract

Within the present NKS NordRisk project means for practical assessment of risks due to long-range atmospheric transport of radionuclides are being developed. Two complementary assessment tools are suggested: 1) An atlas over various dispersion and deposition scenarios based on historical NWP model data coupled to a dispersion model and 2) a simple diffusion model for the ensemble-mean radionuclide activity concentration, combined with a statistical description of the variability around the mean dispersion and deposition values.

Introduction

Loosely speaking, probabilistic risk assessment (PRA) for atmospheric transport addresses the probability that matter released into the atmosphere at a site x ends up at a different position y. More precisely, given a unit release at x, PRA studies aim to quantify the probability distribution of certain risk indicators at y. For radioactive contamination, relevant risk indicators (variables) are the time-integrated activity concentration in air, from which inhalation doses can be derived, and the deposited activity from which external radiation doses or foodstuff contamination levels may be estimated. While the interpretation may be different, PRA applies both to prolonged releases, e.g. from nuclear reprocessing plants or from other industrial activity, and to short term accidental releases of contaminants.

For long-range transport PRA can be obtained through performing numerous and computationally heavy calculations of the atmospheric dispersion and deposition based on numerical weather prediction (NWP) model data coupled to a dispersion model. However, the calculations will be specific to the release site and release characteristics, the physical-chemical properties of the dispersed material, the terrain over which the material is being dispersed, and to the meteorological conditions considered in the study. Hence, even for a single risk site, e.g. a nuclear power plant, when it comes to long-range atmospheric dispersion and deposition it is virtually impossible to cover all likely release and dispersion scenarios.

On the other hand, much simpler, but approximate, risk assessment can be derived based on either historical long-range dispersion data such as the Chernobyl accident release of radio-cesium [1] or from a statistical analysis of a suitably chosen ensemble of long-range atmospheric dispersion model calculations [2]. Within the NKS NordRisk project (Nuclear risk from atmospheric dispersion in Northern Europe) such means for practical assessment of risks due to long-range transport of radionuclides are being developed. Two complementary assessment tools are suggested: 1) An atlas over various dispersion and deposition scenarios, based on an ensemble of (historical) NWP model data coupled to a dispersion model and 2) a simple diffusion model (K-model) for the ensemble-mean radionuclide activity concentration, combined with a statistical description of the variability around the mean dispersion and deposition values.

The risk atlas provides a number of realistic case studies, while specific to the release characteristics and the meteorological conditions considered in the study. General patterns, however, emerge and the gross structure of the observed distributions may be indicative of the dispersion and deposition from other installations or under different meteorological conditions. The K-model further simplifies the description of long-range dispersion but allows for easy tuning of the release and dispersion parameters, e.g. to account for different dry or wet deposition properties of the dispersed matter than assumed in developing the atlas.

Further investigations are still needed to gain insight into the mechanisms underlying the large-scale dispersion patterns and the associated variability before the simplified PRA may become fully operational. Such investigations are undertaken within the present NordRisk II project (2008-9).

Risk Atlas

To develop the risk atlas, two years of analyzed NWP model data from the European Centre for Medium-range Weather Forecast (ECMWF) covering the Northern Hemisphere has been employed [3]. For selected risk sites a hypothetical (unit) release of different radionuclides is assumed. Calculations of the atmospheric dispersion and deposition been carried out with the Danish long-range puff model DERMA [4]. Each puff has been followed for up to three weeks from the time of the release and downwind time-integrated air concentration and dry and wet deposition determined.

The result of the long-range atmospheric transport and deposition is presented in the risk atlas in the form of maps of activity concentrations based one one-year continuous releases. An example of such a map is shown in Fig. 1, giving the ensemble-mean deposition of ^{137}Cs from a hypothetical release from Leningrad NPP. As seen in the figure, the mean-field deposition tends to be rather isotropic, at least for medium-range distances, and the activity concentrations decreases monotonically with the distance from the release site. Some indication of the northeasterly trade winds at low latitudes may be inferred from the figure.

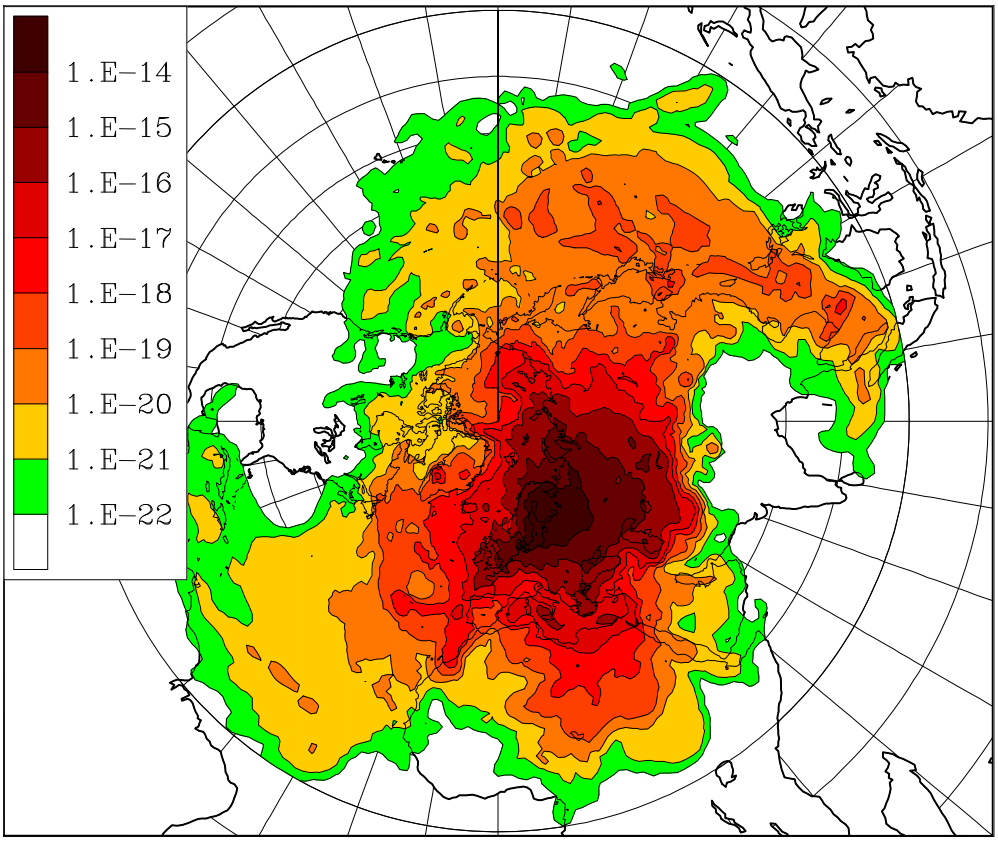

Figure 1. *Ensemble mean deposition of a unit release of ^{137}Cs from Leningrad NPP.*

In all scenarios considered for the atlas, the ensemble-mean depositions show the same general pattern. The deposition length scale (the root-mean-squared distance from the release site to where the material is deposited) varies from a few hundreds kilometers up to a thousand kilometer. The deposition length scale is determined by the large-scale advection velocities and the rate of removal caused by deposition and decay.

For continuous emissions, e.g. from a reprocessing plant, the mean field deposition pattern directly provides the expected geographical scale of the contamination (two-particle correlation length). For very long time scales, one would expect the spatial distribution of the contaminant to approach a mean-field deposition, and the maps may be viewed as approximations to this mean-field. To a first approximation, the concentrations depend only on the distance from the release site. The fluctuations around this mean-field appear random [5] and the fluctuations relative to the isotropic mean-field are well described by a gamma model (Fig. 2).

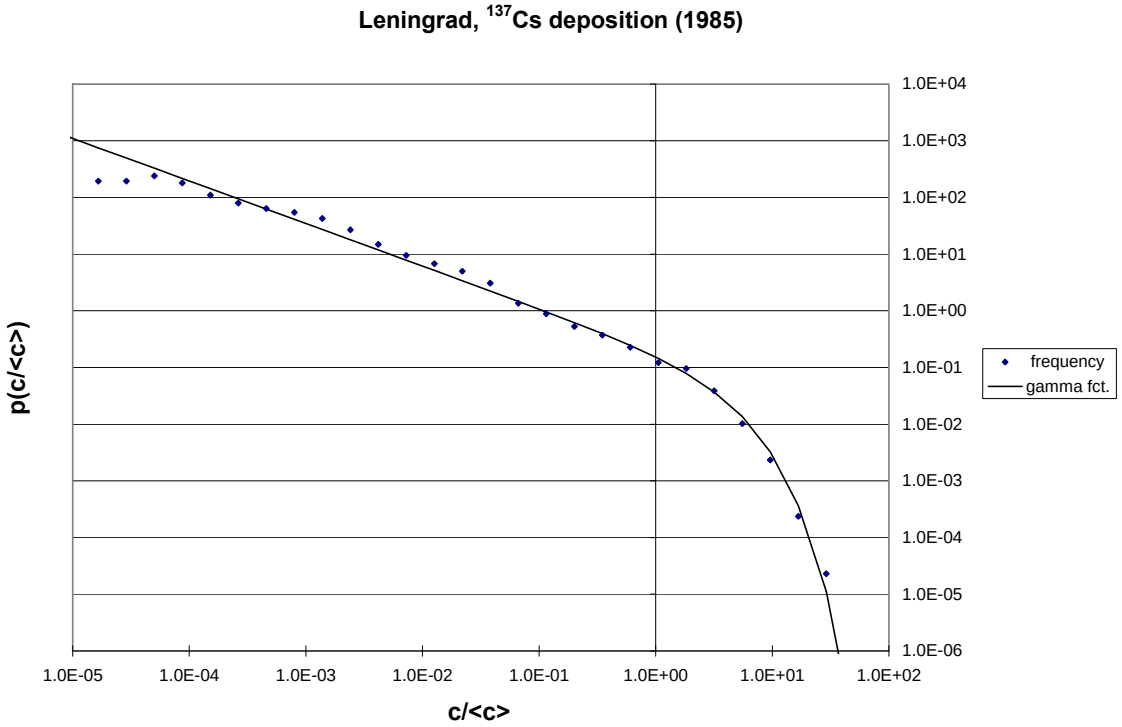

Figure 2. *Probability density function of the relative deposition density* $c/<c>$ *where* $<c>$ *is the (spatially averaged) isotropic mean field. The solid line is a gamma model distribution (of unit mean) with shape parameter* $1/\phi = 0.25$.

Short-term releases

The risk atlas also applies to short-term (accidental) releases but the interpretation in this case is rather different. For a short term release of activity, the plume will not disperse isotropically from the release site, and the two-particle correlation length will be much smaller than the deposition length scale defined above. The ensemble-mean deposition, however, still gives the expectation of the (one-particle) density, but fluctuations around this mean-field value may be large.

Especially for very short releases, very large fluctuations of the concentration field around the ensemble-mean value are expected. The probability density function for the risk indicator may be estimated from the ensemble of dispersion model calculations. Assuming releases of 24 hour duration the distribution of the relative deposition density at a distance of 2000 km from the release site is shown in Fig. 3 for the Leningrad NPP hypothetical release. As for the spatial variability, the probability density function also in this case seems to be well described by a gamma model. The variance, given by the inverse of the shape parameter, however, is much larger than for the case of continuous emissions (Fig. 2.).

The mean-field calculations combined with a statistical model accounting for the variability will constitute a practical probabilistic risk assessment tool based on historical NWP model data. Work is in progress to examine and quantify these fluctuations in more detail.

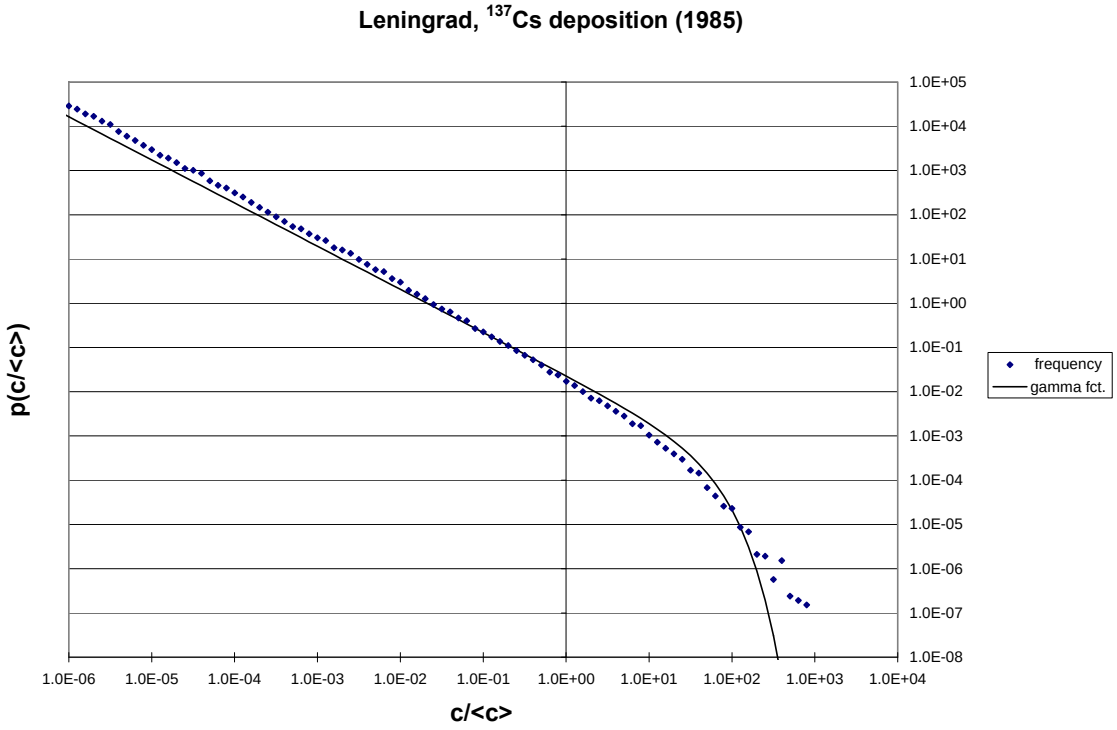

Figure 3. *Probability density function of the relative deposition density, based on 24-hour releases. The solid line is a gamma model distribution with shape parameter* $1/\phi = 0.025$.

K-model approximation

The near-isotropic distribution of the ensemble-mean deposition can be understood in terms of a simplified advection-diffusion model (K-model) with constant diffusivity and deposition parameters. This model may be solved analytically and the results easily coded for graphical presentation. While the model clearly may not account for much of the complexity associated with atmospheric transport and deposition, the model is found, perhaps surprisingly, to capture the gross structure of the ensemble-mean dispersion and deposition [2].

In the NordRisk project, a PC-software tool has been developed for presentation of the K-model deposition fields. The presentation of the results is in the form of maps generated with the web-based Google Earth global graphical display system (http://earth.google.com/). An example of this is shown in Fig. 4. Also the maps developed for the risk atlas are accessible via the Google Earth interface.

The K-model parameters depend on the meteorological conditions and the physical-chemical properties of the dispersed material. The parameters at present must be set externally, but may in principle be inferred from the underlying NWP model data and the DERMA dispersion model parameters. By regression of the activity concentration fields obtained with the K-model against the ensemble-mean fields of the risk atlas, it is found that the K-model parameters depend on the dispersion and deposition properties of the released material, but only to a lesser degree on the coordinates of the release site.

Consequently, the K-model combined with a proper statistical description of the fluctuations may provide a simplified PRA also for risk sites where detailed, long-term numerical atmospheric dispersion model calculations have not been carried out.

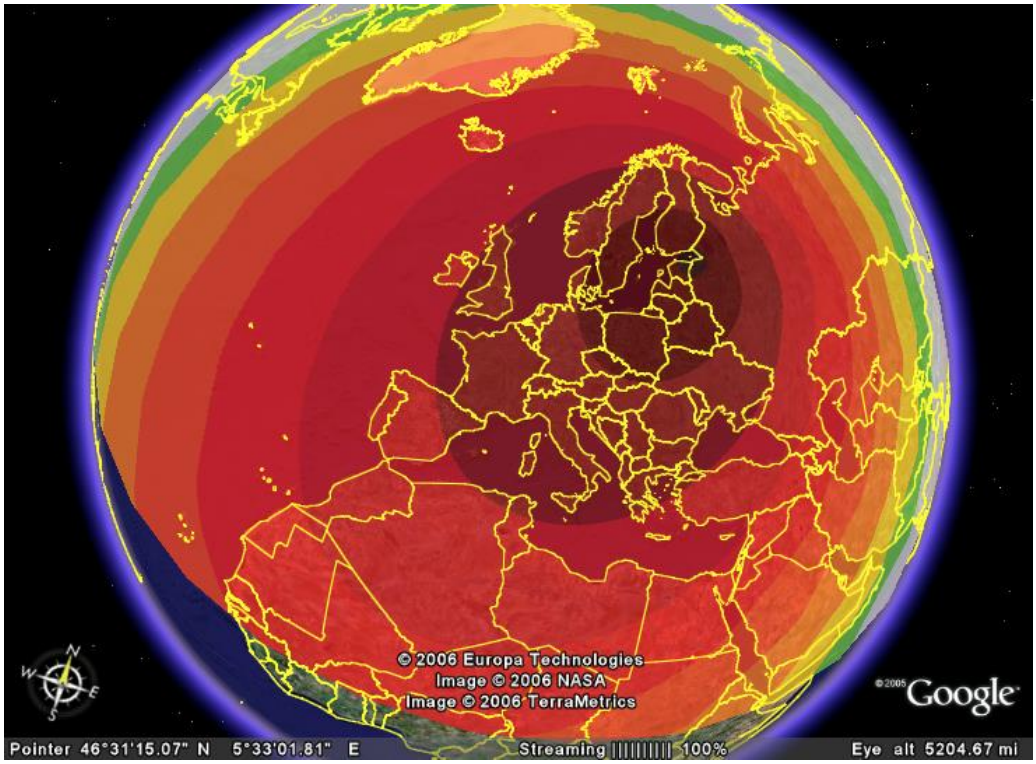

Figure 4. *Simplified dispersion model calculation of the ensemble-mean deposition of ^{137}Cs released from Leningrad NPP. Google Earth image.*

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Design of the KETALE web application to improve collaborative emergency management

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Abstract

KETALE is a database and Web application intended to improve the collaborative decision support of the Finnish Radiation and Nuclear Safety Authority (STUK) and of the Finnish Meteorological Institute (FMI). It integrates distributed modeling (weather forecasts and dispersion predictions by FMI, source term and dose assessments by STUK) and facilitates collaboration and sharing of information. It does so by providing functionality for data acquisition, data management, data visualization, and data analysis.

Introduction

Emergency management of nuclear or radiological disasters is a collaborative effort. The Radiation and Nuclear Safety Authority of Finland, for example, mans in case of exercise or emergency an emergency center, where an accident assessment group analyzes the plant status and makes release assessments, a consequence assessment group analyzes the current and tries to predict the future radiological situation off-site, and a liaison group gives advice to industry and commerce. The whole activity is coordinated by a management group. Evidently, information has to flow seamlessly and timely between all these groups. In addition to this internal information flow, a constant information exchange is needed with other institutions. The off-site consequence assessment group, for example, needs input from the Finnish Meteorological Institute that operates a long-range dispersion model and provides numerical weather prediction data to decision support systems that the off-site consequence assessment group uses. On the other hand, FMI needs the release assessment of STUK for their long-range dispersion model.

The KETALE project was launched to streamline the data and information exchange between STUK and FMI. The goal was to create an application that integrates the distributed modeling applications (weather forecasts and dispersion predictions by FMI, source term and dose assessments by STUK) and facilitates collaboration and sharing of information. This has been achieved by providing functionality for data acquisition, data management, data visualization, and data analysis.

The following design considerations were agreed upon at an early planning stage (see Lahtinen *et al*.):

- Multi-user: The system is typically used simultaneously by many users at many different locations; different users need different views of the system.
- Multi-lingual: It must be possible to produce reports in Finnish, Swedish, and English; a multi-lingual user interface would be an additional asset.

- Ease of use: The system should be easy to use and not require any special or advanced computer skills.
- Few client-side system requirements: There should be only very few client-side requirements.
- Quality controlled: The system must be fit for operational use in an emergency centre, which must be assured by constant quality control.
- Openness: It should be easy to both enter data into and get data out of the system.

Data Acquisition

The data acquisition component is the glue layer between the distributed modeling applications. It implements a simple request-response pattern where a request is issued to an application server, which returns either the calculation results or some fault condition. As most services need considerable time to respond to requests, asynchronous Web service invocations are the preferred choice. However, not all modeling applications implement a Web service interface. In such cases KETALE uses whatever protocol is available to provide input parameters to the model and obtain the results. In the case of FMI's long-rage dispersion model it uses programmatic Web browsing, and to obtain FMI's numerical weather predictions it uses the File Transfer Protocol. In addition, data can also be imported into KETALE by means of simple file upload.

KETALE provides the users with a consistent Web browser interface to all registered models. This bears the advantage that users can access these models without having to care about the particular hardware and software the respective applications are installed. The users do not need to be trained on the use of the sometimes rather scientifically motivated user interfaces of these applications.

Whenever possible, model results are obtained in form of raw data instead of readily rendered images. Raw data can be portrayed by the KETALE users in such a way that best serves their purposes. In addition, raw data can be manipulated in various different ways: it can be scaled, for example, to account for a somewhat higher release than initially thought of; doses from different pathways can be added; the plume arrival time can be calculated from the activity concentration in the air; or results can be compared to intervention levels. Finally, the output data of one model can be fed into another model to obtain additional results.

Data Management

Data management is concerned with keeping account of who was doing what and when. All users have to register before they are granted access to the system. They can have one of four roles: observer, editor, simulator, or administrator. Observers can browse and visualize existing information, but they cannot make changes to the database. Editors, on the other hand, can interact with the system in a way that changes the database. For example, they can add notifications, request model results, add release information, or edit reports. All changes to the database are seen immediately by other users. Simulators have editor privileges with the difference that their datasets are not seen by other users. This role can be used to prepare exercises. Finally, to remove datasets, or to add and remove other users, one needs administrator privileges.

A relational database is used to persistently store all data objects, be it metadata related to events, sites, requests, responses, and reports; or release information or any other information in the form of free

text. In addition, all imported data is registered in the KETALE database. The necessary metadata is either extracted from the response, or it can be supplied manually during import.

Users are presented with a notification table (Figure 1) that shows all recent database transactions (emergency declaration, source term input, issue of a request, receipt of a response, and addition of a report). By clicking on an entry they can view the notification in detail and, in the case of a model result, visualize it. Filters are provided so that users can list only entries of interest.

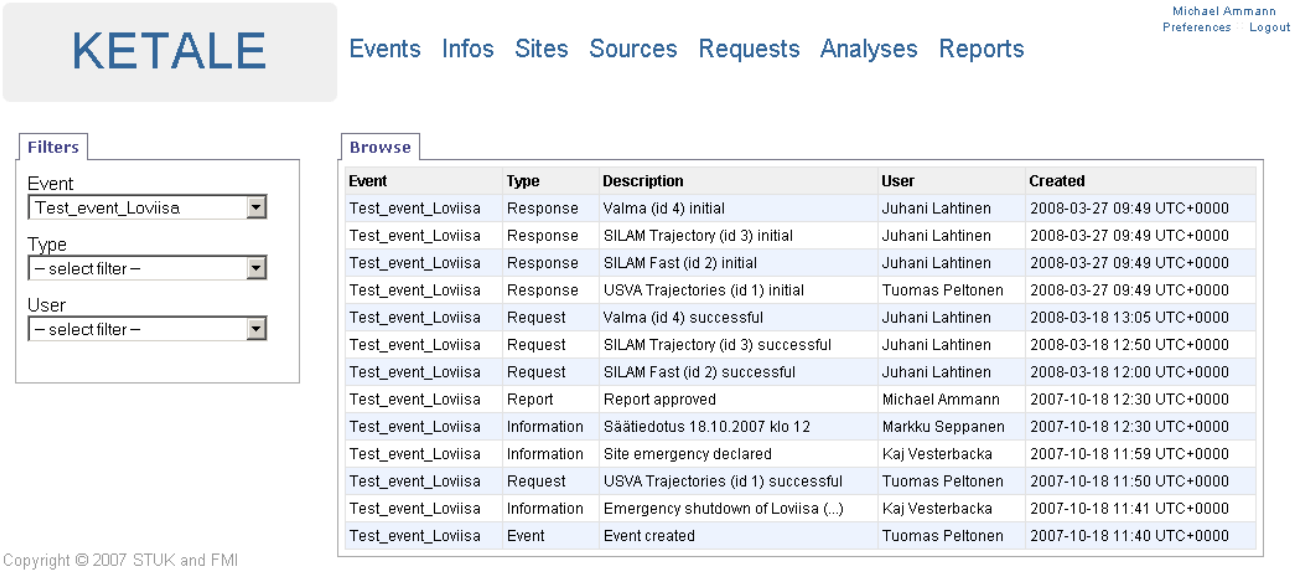

Figure 1. *Notification page*

Data Visualization

The data visualization page consists of a Web GIS component and of control widgets (Figure 2). An action widget allows the user to change style, scale, color map, and map projection of the data portrayal. Users can zoom and pan to the area of interest, and add or replace underlying map layers. The customized map can be annotated and added to a report that can be later exported as a PDF document. Selection widgets are available if the data has time and height dimensions. The evolution over time can be animated, or the respective time function pertaining at any location can be displayed just by clicking on the map.

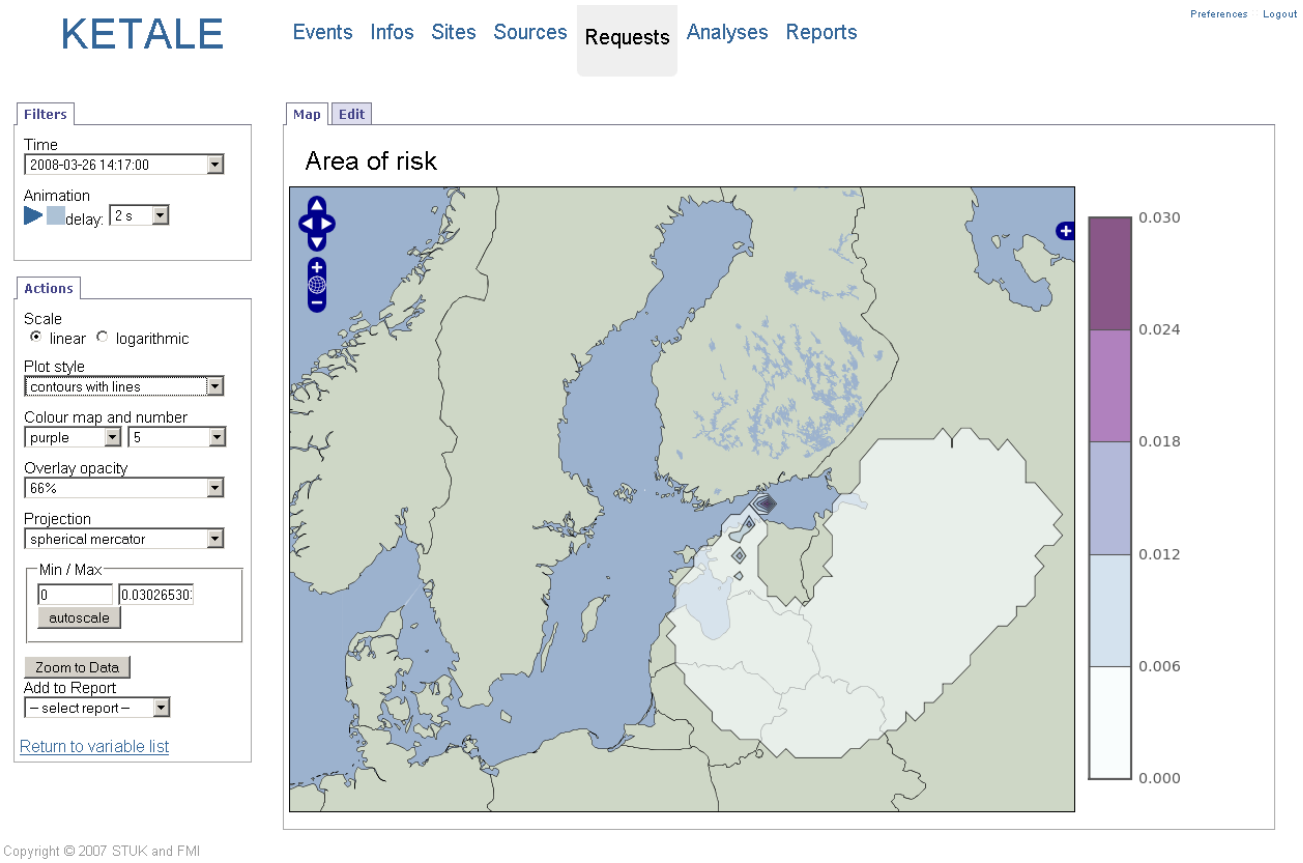

Figure 2. *Visualization page (hypothetical scenario)*

System implementation

KETALE was developed using the TurboGears Web application framework. The design follows the Model-View-Controller (MVC) design pattern. This is a popular design pattern in Web applications encouraging modularity as it separates database (model), application logic (controller), and user interface (view). The user interface is expressed in a templating language (Genshi) with the addition of JavaScript (MochiKit). It is dynamically populated during runtime with data from the database. Ajax technology is used to make the application more responsive by exchanging data with the Web server behind the scenes, instead of reloading an entire Web page each time a user makes a change. The data objects are linked to a relational database management system (MySQL) with an object relational mapper (SQLAlchemy). This bears the advantage of being independent of a particular database system and of easier coding. Model results are mostly stored using NetCDF, which is a set of software libraries and machine-independent data formats that support the creation, access, and sharing of array-oriented scientific data.

The application is written in Python and uses the CherryPy Web server. Python is a general purpose scripting language that is well suited for gluing together the different components of a system of this kind. CherryPy is an easy-to-use, object-oriented Web server, which allows developers to build Web applications in more or less the same way they would build any other object-oriented Python program.

Maps are displayed using the OpenLayers library. OpenLayers is a JavaScript API for displaying map data in Web browsers and adds functionality for navigation and zooming. It implements industry-

standard methods for geographic data access, such as the OpenGIS Consortium's Web Mapping Service (WMS) and Web Feature Service (WFS) protocols. This makes it easy to switch between different map services without demanding changes to the code. The map in the Figure 2, for example, is produced by STUK's own WMS that uses the GeoServer software. The data layer on the map is requested with a WMS-like call with extra parameters.

There are many Web application frameworks available today that support the development of dynamic Web applications. Existing experience and personal taste are probably the major considerations in deciding on a particular toolset.

Conclusions

Modeling applications often unduly impose their requirements on the users: applications have to be installed on particular hardware; users have to get acquainted with the operation system of that specific hardware and with the application logic of the model. Furthermore, they have to content themselves with predefined and often static results. KETALE tries to correct that. A modern Web browser is all that is needed to access and interact with the system. The users can access the platform independent system virtually from anywhere and they are confronted with a consistent user interface that they can use to access a multitude of application programs.

The system can be (and is typically) used simultaneously by many users with various needs. All of them are provided with their own view of the data. In addition, the system is multi-lingual. Finnish, Swedish, and English are the currently supported languages. Users can select their default languages and switch to another language whenever they want.

Finally, emergency management demands security and operational reliability. Security is provided by the secure socket layer (SSL) and, in the future, by a dedicated governmental backbone network, whereas the operational fitness is achieved by constant quality control.

In conclusion, we want to highlight the following system properties:

- Web application: The system can be accessed with a modern Web browser over the Internet, and no client-side installation is needed.
- Database centric: All information is stored in a central database; users are provided with context-specific views of the same database.
- Database agnostic: The system does not depend on a particular database management system.
- Modular: The model-controller-view design pattern naturally separates the various parts of the system and enforces modularity.
- Web map visualization: Model results can be shown on top of any geographic map that is served by a Web Map Service; basic navigation tools are provided.
- Open source software: The system is build completely with freely available open source software.
- Platform independent: The system can be deployed both on Linux or Windows platforms; users access the system from a Web browser interface that has no client-side requirements.

Acknowledgement

We highly appreciated the contribution of Eero Kurkela to the initial version of KETALE in 2007. He was then a student at the Helsinki University of Technology.

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A European network of experts directly responsible for monitoring and assessment of doses after deliberate exposure to ionizing radiation

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Abstract

In the event of an accidental or deliberate release of radionuclides to the environment, individual monitoring and dose assessment may be needed for large numbers of people. The consequences of such incidents are not limited by national boundaries. However, within the European Union (EU), there is no coordinated strategy for individual monitoring and dose assessment and little exchange of information. In the event of an accidental release of radionuclides (e.g. a reactor accident) or a deliberate release (e.g. a terrorist incident involving a radiological dispersion device (RDD)), individual monitoring and dose assessment may be needed for large numbers of people to identify those needing medical treatment, and to inform and reassure others. The affected population would include both members of the public and emergency workers responding to the incident.

CONRAD (**CO**-ordinatio**N** Action for **RA**diation **D**osimetry) is an EC 6th Framework Programme Co-ordination Action sponsored by EURADOS (the European Radiation Dosimetry Group, http://www.eurados.org). Task 5.4 of the CONRAD project was to establish the EUREMON network, which aims to promote sharing of information between countries on plans and arrangements for individual monitoring.

The network currently has 51 individual members from 22 EU countries, 8 non-EU countries and two international organisations.

Introduction

In the event of an accidental or deliberate release of radionuclides to the environment, individual monitoring and dose assessment may be needed for large numbers of people. Those needing medical treatment, probably a much smaller number, have to be identified and others need to be informed and reassured. The affected population would include both members of the public and emergency workers responding to the incident. The event could be an accidental release of radionuclides (e.g. a reactor accident) or a deliberate release (e.g. a terrorist incident involving a radiological dispersion device (RDD)). The consequences of such incidents are not limited by national boundaries. However, within

the European Union (EU), there is no coordinated strategy for individual monitoring and dose assessment and little exchange of information.

CONRAD (**CO**-ordinatio**N** Action for **RA**diation **D**osimetry) is an EC 6th Framework Programme Co-ordination Action in EURADOS (**EU**ropean **RA**diation **DOS**imetry Group (http://www.eurados.org)). Work Package 5 (WP5) within CONRAD dealt with the coordination of research on internal dosimetry. Task 5.4 of CONRAD WP5 was to establish the European Emergency Monitoring network, which aims to promote sharing of information between countries on plans and arrangements for individual monitoring. Simultaneously there is a EC project TMT Handbook -Triage, Monitoring and Treatment of people after malevolent exposure to ionizing radiation.

Establishing the network

The network was given the acronym EUREMON: **EUR**opean **E**mergency **MON**itoring. It was decided that the network should include people with direct responsibility for monitoring and dosimetry of people affected as a result of an incident involving either the accidental release or the malevolent use of radiation or radioactive materials. Invitations were sent to selected people in Europe known to be active in the area of interest. The stated aims of the network are to collect and share information on: relevant national legislation; any international agreements or existing collaborations relating to monitoring and dose assessment; publications & reports describing strategies for internal and external contamination monitoring; publications & reports giving advice on therapy; equipment for internal/external contamination monitoring after accidental or deliberate releases; current dose assessment methodologies; and currently-available dose assessment software.

Survey of current practice

The survey of current practice was carried out by means of a questionnaire which was developed in such a way that it would meet the needs of both the CONRAD and TMT Handbook projects. It was decided to ask a limited number of questions, but to make them open questions so the recipient would have freedom to give more or less information on a particular topic, as appropriate. To make clear the type of material required, example responses to each question were provided within the body of the questionnaire. Recipients were invited to reply by providing references or copies of existing published documents, and/or by providing a brief written response to each question.

European Radiation Dosimetry Group

EURADOS

Health Protection Agency

TMT Handbook

STUK Radiation and Nuclear Safety Authority

Monitoring, management and treatment of members of the public and emergency workers after accidental or deliberate releases of radionuclides or exposure to radiation

1.1 Purpose of this questionnaire

This is a request for information on the current status of arrangements for the triage [1], monitoring [2] and treatment of members of the public and emergency workers after accidental or deliberate [3] releases of radionuclides or exposure to radiation. Topics of interest to us include national strategic plans, monitoring strategies, availability of monitoring resources, capabilities for the interpretation of monitoring data, recommendations on medical management, relevant national emergency exercises, and research and development projects aimed at improving capabilities in this area.

1.2 How the information will be used

The information will be used in two 6th Framework Programme projects part-funded by the European Commission:

> TMT HANDBOOK (http://www.tmthandbook.org) will provide a practicable handbook for the effective and timely Triage, Monitoring and Treatment (TMT) of people following a malevolent act. This is a Specific Targeted Research Project. (Contract no: FP6 - 036497)
>
> CONRAD (Task 5.4) aims to collect and share information on the topics listed above, and also aims to define needs for research, advice and training. This is a Co-ordination Action sponsored by EURADOS (European Radiation Dosimetry Group). (Contract no: FP6 - 12684)

The information you provide will make a valuable contribution to both projects. Individuals, organisations or countries providing information will not be identified in any reports or publications issued unless permission is explicitly given.

What to do now

You only need to answer those questions that are within your area of expertise. Please enter your responses in the fields shaded in light grey. The preferred language is English. Please return the completed questionnaire by e-mail to: peter.pellow@hpa.org.uk. Alternatively, you may fax it to the Health Protection Agency, for the attention of Dr P G D Pellow (++44 1235 833891).

Thank you for your assistance.

Explanatory notes

1. "Triage" – the use of simple procedures to prioritise people rapidly for further actions, including monitoring and treatment, in order to maximise the effective use of resources.

2. "Monitoring" – measurements of external radiation fields, external contamination or internal contamination; and the interpretation of such measurements in terms of radiation doses to the individual. The use of cytogenetics for biological dosimetry is included. The various types of monitoring are referred to generically as "emergency personal monitoring".

3. That is, the malevolent use of radioactive material in a public place.

Table 1 Organisations participating in CONRAD Task 5.4

Organisation	Country	Full / Corresponding Member	Participants	Notes
HPA	United Kingdom	Full	George Etherington	
			Peter Pellow	From January 2006
			Alan Hodgson	Till January 2006
			Neil Stradling	
STUK	Finland	Full	Tua Rahola	
			Maarit Muikku	
ENEA	Italy	Corresponding	Paolo Battisti	
CEA	France	Corresponding	Philippe Bérard	
IRSN	France	Corresponding	Cecile Challeton-de-Vathaire	
EdF	France	Corresponding	Bernard Le Guen	
NRPA	Czech Republic	Corresponding	Irena Malatova	

Table 2 TMT HANDBOOK participants contributing to the development of the questionnaire

Name of Organisation	Abbreviation	Organisation location
Belgian Nuclear Research Centre	SCK•CEN	Belgium
Norwegian Radiation Protection Authority	NRPA	Norway
Enviros Consulting Ltd.	Enviros	United Kingdom
Radiation and Nuclear Safety Authority	STUK	Finland
Health Protection Agency	HPA	United Kingdom
World Health Organisation	WHO	Geneva
Central Laboratory for Radiological Protection	CLRP	Poland

Section A requests the name and address of the person completing the questionnaire and the organisation to which they are affiliated.

Section B asks the person replying to provide details (i.e. name, address, organisation and expertise) of people within their country who would have direct responsibility for monitoring or management of those people exposed to radiation as a result of an incident; and references and/or internet addresses (uniform resource locators (URL)) for any existing networks relating to the monitoring or management of people after an exposure.

Sections C-L requests information, including references to supporting papers or electronic documents where available, on both theoretical and practical topics that would need to be addressed in dealing with a radiation incident. Where appropriate, a brief description and/or examples of the type of information requested are included at the beginning of the section to help the responder.

The section headings for C-L are:

Section C: Organisations and responsibilities

Section D: Relevant legislation, guidance, international agreements, etc

Section E: Recommendations on monitoring strategies

Section F: Equipment and facilities for emergency personal monitoring

Section G: Recommendations on methods for emergency personal monitoring

Section H: Dose assessment methodologies and software

Section I: Recommendations on medical management

Section J: Public information and communication

Section K: Emergency exercises

Section L: Research and development projects

Finally, Section M asks the responder to provide any other information that they consider relevant.

The questionnaire was sent to:

- Contributors to a survey on monitoring practice carried out for an earlier project (the 5th Framework Programme project OMINEX)
- EURADOS members
- Participants in Work Package 5 of CONRAD
- Members of EURADOS's harmonisation network
- People known to be active in the area of interest to Task 5.4

The network currently has 51 individual members from 30 European states (22 EU countries, 8 non-EU countries) and two international organisations. Three E-mails did not reach destination, two responders were unable to help, 19 questionnaires were returned and 27 members did not reply at all. This is a slight under-estimate of the true return rate because in some instances questionnaires were sent to more than one person within an organisation in a particular country whereas a single questionnaire was returned for that organisation, e.g. STUK (Finland), GAEC (Greece) and NRPA (Norway).

Compilation of information on portable and transportable monitoring facilities

Information has also been collected on portable and transportable monitoring facilities and equipment that could be used for individual monitoring in the event of an accidental or deliberate release of radioactive material resulting in internal or external contamination. Here "portable" means equipment that can be carried and operated by hand by an individual or a small team, while "transportable" means equipment that must be transported by a vehicle (e.g. a car, truck, train or aircraft). The aim is to compile information that can be madefreely available to national authorities who may require assistance in carrying out individual monitoring of members of the public in the event of an emergency.

To date, information has been compiled in the form of short reports for four EU countries: Finland, France, Spain and the United Kingdom. These reports are appended to this report in Appendices E, F, G and H. Other EUREMON members will be invited to contribute similar reports. The compilation will then be published in the open scientific literature.

Conclusions

A network of people with direct responsibility for monitoring and dosimetry of people in the event of a radiological emergency has been established. The following possible issues that would promote emergency preparedness in the field of individual monitoring and dose assessment were identified:

- The need for national surveys of inventories of available equipment suitable for emergency personal monitoring
- The development of desktop emergency exercises as a training tool for EUREMON members, focussed on monitoring, dose assessment, and management of members of the public
- The need for guidance on rapid in vitro analysis methods
- The need for dose assessment software appropriate for exposures received by members of the public, including assessment of absorbed doses received over short periods
- The need for guidance on the use of Prussian Blue for enhancing the clearance of radiocaesium, aimed at limiting exposures of members of the public.
- The promotion of cooperation and sharing of resources (e.g. bioassay laboratories) between EU countries

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Figure 1. *STUK's new vehicle for whole-body counting.*

Figure 2. *IRSN's new vehicle for whole-body counting.*

Norwegian Assessment of Nuclear and Radiological Threats

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Abstract

Even though the Norwegian nuclear industry is limited, it has since the Chernobyl accident in 1986 been Norwegian policy to have a particular emergency preparedness towards nuclear or radiological events. In order to maintain an effective emergency preparedness with limited resources, it is necessary to have a good understanding of possible scenarios and related consequences. Norwegian authorities are therefore continuously assessing nuclear and radiological threats.

Due to their close vicinity to the Norwegian border and poor conditions at some of the sites, possible accidents at nuclear installations on the Kola Peninsula in Russia have been emphasised in Norwegian emergency preparedness. However, Norwegian and international efforts during the last few years have contributed to improve the conditions at several sites on the Kola Peninsula. Improved safety and physical protection at these sites have reduced the risk for cross-border contamination. Furthermore, extensive decommissioning and dismantlement of aged nuclear submarines have reduced the number of nuclear objects of particular concern.

Meanwhile, the latest changes in the international security environment have made one consider a new set of scenarios, involving e.g. massive attacks on nuclear installations. Aging nuclear installations, proliferation of nuclear weaponry and an international nuclear power renaissance have also contributed to change the Norwegian threat perception.

The consequences of the tsunami in East Asia in December 2004 has lead to an increased understanding of the authorities' responsibilities concerning Norwegian citizens and interests abroad, also with respect to nuclear and radiological events that will not bear any impact on Norwegian territory. In particular, the increasing attention towards the economic potential in the Russian High North implies new challenges regarding an increased Norwegian establishment in an area with still many problems concerning radioactivity.

Finally, awareness is also directed towards future climate changes and challenges related to nuclear safety and radioactive contamination.

NKS-B: On-going activities and future priority areas for research

Justin Gwynn

NKS-B Program Manager, NKS (Nordic Nuclear Safety Research)

With a total population of some 25 million people, and a common cultural and historic heritage, the Nordic countries have cooperated in the field of nuclear safety for approximately half a century. Informal networks for exchange of information have developed throughout the years, strengthening the region's potential for fast, coordinated and adequate responses to nuclear threats, incidents and accidents. NKS (Nordic Nuclear Safety Research) is a platform for Nordic cooperation and competence in nuclear safety, including radiation protection and emergency preparedness. The work is centred on nuclear power related issues and is divided into two main areas: Reactor safety (NKS-R) and emergency preparedness and environmental issues (NKS-B). The purpose of NKS is to carry out activities producing seminars, exercises, scientific articles, technical reports and other types of reference material, while special efforts are made to engage young scientists in this work. Activities are financed and supported by the Nordic authorities, research institutions, power companies, contractors and other organizations.

Owners and Main Financiers

DK Danish Emergency Management Agency

FI Ministry of Industry and Trade

IS Islandic Radiation Protection Institute

NO Norwegian Radiation Protection Authority

SE The Swedish Nuclear Power Inspectorate

The Swedish Radiation Protection Authority

(Swedish Radiation Safety Authority)

Additional funding from:

• Fortum Power and Heat Oy (Finland)

• TVO (Finland)

• IFE (Norway)

• Forsmarks Kraftgrupp AB (Sweden)

• Kärnsäkerhet och utbildning (KSU) AB (Sweden)

• OKG Aktiebolag (Sweden)

• Ringhals AB (Sweden)

The aim of the NKS-B program is to strengthen Nordic competence in the fields of radiological emergency preparedness, measurement strategy, technology and quality assurance, radioecology and the management of radioactive waste and discharges.

NKS-B activities in 2008

PARDNOR

Emergency preparedness

Co-ordinator - Sven P Nielsen (Denmark)

Partners - Finland, Iceland, Norway, Sweden, Faroe Islands

Ongoing activity from 2007 where previously PARDNOR generated up-to-date and Nordic relevant parameters to improve calculated ingestion doses via the ECOSYS model which is used in the decision support systems ARGOS and RODOS. In 2008, PARDNOR will seek to determine Nordic soil migration parameters (leaching, fixation, desorption rates) and Nordic leaf area indices (a critical parameter in determining dry and wet deposition as well as performing sensitivity analyses of some of these parameters in order to understand their associated variance.

GAPRAD

Radiological assessment

Co-ordinator – Justin Brown (Norway)

Partners - Denmark, Finland, Sweden

Ongoing activity from 2007 with the goal of improving the understanding of natural background doses to reference organisms. In 2008, GAPRAD will focus on ^{210}Po and other natural radionuclide transfer in terrestrial and freshwater environments and natural radionuclides in the Baltic Sea.

BIOPEX

Emergency preparedness

Co-ordinator – Carlita Lindholm (Finland)

Partners - Norway, Sweden

BIOPEX builds on previous the NKS-B activity BIODOS, where a PCC (Prematurely condensed chromosome) method was established as a rapid biodosimetry assay for emergency preparedness and compared against the standard dicentric biodosimetry assay. In 2008, BIOPEX will conduct an accident scenario exercise to test the established PCC assay, enabling the partcipants to gain experience in handling large numbers of samples and processing the resulting data.

GammaRate

Measurement strategy, technology + QA

Co-ordinator – Hans Bjerke (Norway)

Partners - Denmark, Finland, Iceland, Sweden

A new activity, which aims to harmonize calibration of handheld dosemeters used in emergency situations and propose guidance documents for use and maintenance of dosemeters (for emergency

personnel). In 2008, GammaRate will organise a workshop which will seek to draw upon experience gained from previous emergency preparedness exercises such as DEMOEX.

REMSPEC
Emergency preparedness

Co-ordinator – Mark Dowdall (Norway)

Partners - Denmark, Finland, Iceland, Norway, Sweden

A new activity, which is centered on an exercise involving the distribution and analysis of spectral data intended to simulate the sort of signal an analyst may expect to encounter after a nuclear event. Participants will gain experience from the practice of having to analyse complex spectra under time constraints, with an initial assessment required after a matter hours, to be followed by a more detailed assessment a week after the receiving the spectra. The exercise will also give participants the practice in analysing data not drawn from they own instrumental setup.

DepEstimate
Emergency preparedness

Co-ordinator – Sigurður Emil Pálsson (Iceland)

Partners - Denmark, Finland

A new activity that builds upon previous work showing that precipitation can be used to successfully estimate fallout deposition, while recognising the fact that better deposition estimates can be obtained if precipitation concentration time series are available in addition to precipitation time series. DepEstimate seeks to validate the precipitation concentration time series function derived using global data sets with Nordic data, before attempting to expand the function to include Pu isotopes combined with validation of this method of estimating fallout Pu deposition whilst taking other sources of Pu into account.

FOREST-2
Emergency preparedness

Co-ordinator – Virve Vetikko (Finland)

Partners – Finland, Norway, Sweden

A new activity, but closely related to the previous NKS-B FOREST activity, the aim of FOREST-2 will be to hold a seminar in Helsinki in October 2008 as a forum for scientific communication to Nordic forest radioecologists and forestry experts, providing information about radioactive contamination of forests and mitigation of the effects of contamination. In addition, the FOREST sampling guide produced during the earlier NKS-B FOREST activity will be demonstrated.

SPECIATION
Measurement strategy, technology + QA

Co-ordinator – Xiaolin Hou (Denmark)

Partners - Finland, Norway, Sweden + Lithuania

Ongoing activity from 2007, SPECIATION in 2008 will seek to further develop and improve analytical methods for speciation of several important radionuclides in the environment (Pu, Am, Np, ^{137}Cs, ^{90}Sr, ^{129}I, and ^{99}Tc) and optimize and harmonize separation techniques for radionuclide speciation that have been developed in Nordic laboratories. Work in 2008 will include an intercomparison on the speciation analysis in soils and sediments as well as targeting radionuclide speciation in specific Nordic environments such as Pu in Thule and radionuclides in Nordic waters resulting from discharges from European reprocessing facilities.

HAIRPOL

Radiological assessment

Co-ordinator – Elis Holm (Sweden)

Partners - Denmark, Norway

A new activity that will seek to provide data on ^{210}Po and ^{210}Pb transfer coefficients to hair and feathers for man and animals and investigate the possibility of using hair as an indicator of ^{210}Po and ^{210}Pb body burdens and intake instead of urine and faeces. In addition, HAIRPOL will test the hypothesis that it is sulphur in the hair and feathers responsible for the accumulation of these radionuclides and that polonium and lead in hair and feathers reflects mainly oral intake and not external contamination. In doing so, HAIPOL aims to improve on existing radioanalytical techniques for ^{210}Po and ^{210}Pb.

Further details concerning these activities and information on entire NKS program, including opportunities for funding can be found on the NKS website www.nks.org.

Chewing gum for retrospective determinations of radiation doses, by means of EPR analysis

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Abstract

In case of an accident or attack involving radiation there is need for fast and accurate estimations of absorbed dose to individuals to optimize the medical treatment or in case of moderate doses to estimate the risk of late effects.

Ionizing radiation induces free radicals which are rather stable especially in crystalline materials. By means of EPR (Electron Paramagnetic Resonance) spectroscopy the number of free radicals can be measured, which is proportional to the absorbed dose in the material. EPR-dosimetry on tooth enamel is today one of the best method as a rather fast method for dose reconstructions. Till now a drawback is that accurate analysis must be done at extracted teeth.

In the pockets of individuals some materials like sweets, medicines and chewing gum has a content of sugar or sweetening which can be used for EPR-dosimetry. After a brief testing sucrose and sugar free chewing gums were chosen and the specific aim was to analyze the EPR spectra, establish a dose response curve and estimate the accuracy for dose reconstructions.

Samples of 300 mg of sucrose were put into test tubes and irradiated with doses from 0 to 21 Gy and 10 chewing gums were irradiated from 1 to10 Gy. A regression curve was obtained for both measurements and the accuracy of the dose reconstruction was checked in a blind test with irradiation of a chewing gum. A dose of 4,24±0,35Gy (1SD) was estimated for the blind dose, which true value was 3,96 Gy. Further investigations are believed to improve the accuracy significantly.

Introduction

In case of an accident or attack involving ionizing radiation there is need for fast and accurate estimations of absorbed dose to individuals to optimize the medical treatment or in case of moderate doses to estimate the risk of late effects.

Ionizing radiation induces the creation of radicals in irradiated matter. Radicals are normally very short-lived, but can remain almost stable in some crystalline materials. By means of EPR (Electron Paramagnetic Resonance) spectroscopy the number of free radicals can be measured, which is proportional to the absorbed dose in the material. EPR dosimetry on tooth enamel is today one of the best methods for retrospective dosimetry (Romanyukha et al. 2000). Till now a drawback has been that accurate analysis must be done at extracted teeth, but recently tooth doses has been determined in-vivo (Schwartz et al. 2006). Other body materials have been tested like bone, fingernails and hair. Bone is a good material but the analysis must be performed at biopsies, thus this method is rather invasive. Finger nails have been tested (Steen 2006, Chandra and Symons 1987) and the possibilities should be further investigated. Because of very large individual differences with a high variation in EPR signal the use of hair seem to be impossible for retrospective dosimetry using EPR. Sugar has been used for accident dosimetry and is a promising material for dose determinations (Sagstuen 1983).

In the pockets of individuals some materials like sweets, medicines and chewing gum has a content of sugar or sweetening which can be used for EPR dosimetry. After a brief testing sucrose- and sugar-free chewing gums were chosen, especially the brand V6. The specific aim was to analyze the EPR spectra, establish dose response curves, and estimate the lowest detectable dose and the accuracy for dose reconstructions.

Materials and methods

The investigated materials were granulated sugar (sucrose) and the chewing gums V6[1], Stimorol[2], Juicy Fruit[3] and Extra[4]. Juice Fruit contains sucrose and the others contain different types of sweetenings like xylitol, aspartate etc. V6 was chosen for a more detailed study. The specifications of V6 are: Dimensions (maximum): 1.8 cm · 1.4 cm · 0.6 cm, volume: 1cm^3, mass: 1.4 g.

400 mg of granulated sugar, sucrose, was filled into tubes of quartz with an outer diameter of 5.0 mm and an inner of 3.3 mm. The tube was closed with at 5 mm layer of paraffin. To insert the chewing gum into the test tubes, the gums had to be crushed. Since they were sticky they were frozen to 77 K with liquid nitrogen and crushed with a hammer to obtain small grains (< 2 mm). For all types of chewing gums, except V6, these small grains were put into the same kind of quartz tubes as used for sucrose. For V6, tablets were prepared using a tablet press, which formed the crushed grains into cylindrical tablets with the height of 10 mm, diameter of 5 mm and a mass of 200 mg.

All samples were irradiated at room temperature with a 6 MV linear accelerator, Varian Clinac 600 C/D. The mean dose rate was 3 Gy/min. The given doses were determined as dose in water and given in Monitor Units (MU). 1 MU = 0.01 Gy. The Source Sample Distance (SSD) was 100 cm and the field size 10 cm · 10 cm. A tissue equivalent bolus of 1.5 cm was used as a build-up material. The irradiated part of the quartz tubes was never inserted in the spectrometer cavity to avoid EPR signals from the quartz. In the case of V6, whole chewing gums were irradiated before crushed into grains. During irradiation they were placed in a circle with a diameter of about 10 cm were the radiation field is homogenous. In order to shorten the total irradiation time the samples received doses additionally, which means that the samples one by one were removed from the radiation field as a new dose was given. The V6 chewing gums were irradiated to a maximum of 10 Gy in steps of 1 Gy.

EPR analysis

Sucrose: 5 samples, irradiated to doses from 1 Gy to 14 Gy plus one unirradiated, were analyzed. The EPR analysis was performed not earlier than 80 hours after irradiation to avoid signal build-up from the samples as reported by Desrosier (2005). Each sample was measured four times with 4 sweeps of 4 minutes. The field width was ± 5 mT, the time constant 0.1 s. The modulation amplitude was 1.25 mT. To compensate for slight changes in spectrometer sensitivity a Mn^{2+}/MgO reference sample was inserted in the spectrometer cavity. The obtained spectra were analyzed using a computer code in

[1] Dental V6®, Cadbury Sweden AB, [2]Stimorol®, Cadbury Sweden AB, [3]Juicy Fruit®, Wrigley Scandinavia AB
[4]Extra®, Wrigley Scandinavia AB

MatLab. (Israelsson 2008) The spectra were normalized using one of the Mn^{2+}/MgO reference peaks. The peak-to-peak values were obtained for all samples and plotted as dose response curves.

Stimorol, Juicy Fruit and Extra: The EPR spectra were obtained for unirradiated gums and those irradiated with 20 Gy. The modulation amplitude was 0.5 mT and no Mn^{2+}/MgO reference was used. Each sample was measured through 4 sweeps of 4 minutes. Sweep width was 7.5 mT and time constant 0.1 s.

V6: Ten V6 chewing gums were irradiated to 1, 2, 3…8, 9, 10 Gy and one was kept unirradiated. Each gum was prepared into 3 tablets. Each tablet was measured through 4 sweeps of 30 seconds with a modulation amplitude of 0.5 mT, sweep width 5 mT and time constant 0.01 s. The tablets were analyzed 7 days after exposure and 12 days after exposure respectively. For the purpose of noise reduction the spectra were smoothed with the spectrometer software filtration program. Spectra were then normalized to one of the Mn^{2+}/MgO reference peaks. They were also corrected for having base line slopes by subtraction of a linearly fitted base line.

In order to investigate any change in signal intensity from the measurements 7 and 12 days after exposure, all background subtracted spectra were integrated to obtain the absorption spectra. The absorption spectra were then corrected for different base line slopes and the Full Width at Half Maximum (FWHM) for the doses from 5 Gy to 10 Gy were measured.

One additional V6 chewing gum was irradiated to a dose between 0 and 6 Gy unknown to the experimentalist. This gum was prepared into six tablets, which were each measured with the same parameters as the other 10 gums. To calculate the unknown dose the peak-to-peak values from these six tablets were determined and the mean value was used as the measured blind dose.

The obtained peak-to-peak values for each investigated material were assumed to be linearly dependent of the absorbed dose. A regression analysis was performed according to Blom (1989).

Results and Discussion

Typical spectra of sucrose obtained with a modulation amplitude of 1.25 mT are shown in figure 1 for the irradiation doses 0, 8, 14 and 20 Gy. Because of the high modulation amplitude (used for improved accuracy in dose determinations) no details in the spectra are visible. The corresponding dose response is shown in figure 2.

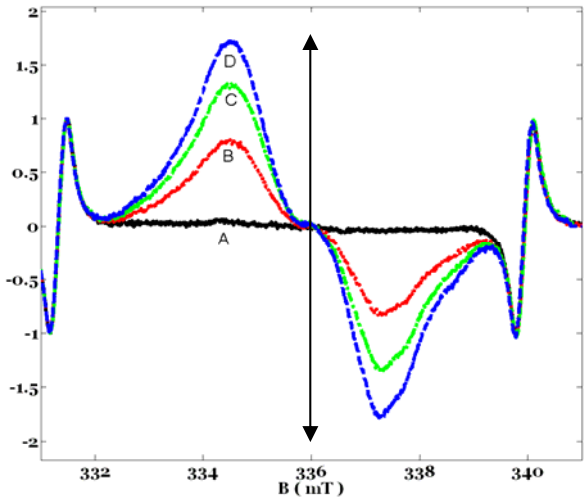

Figure 1. *The EPR signal for sucrose as a function of applied magnetic field. Modulation amplitud is 1,25 mT. Given doses are A = 0 Gy, B = 8 Gy, C = 14 Gy and D = 20 Gy. The arrow shows the peak-to peak estimation for curve D. The signals at 331 and 340 mT are due to the Mn reference.*

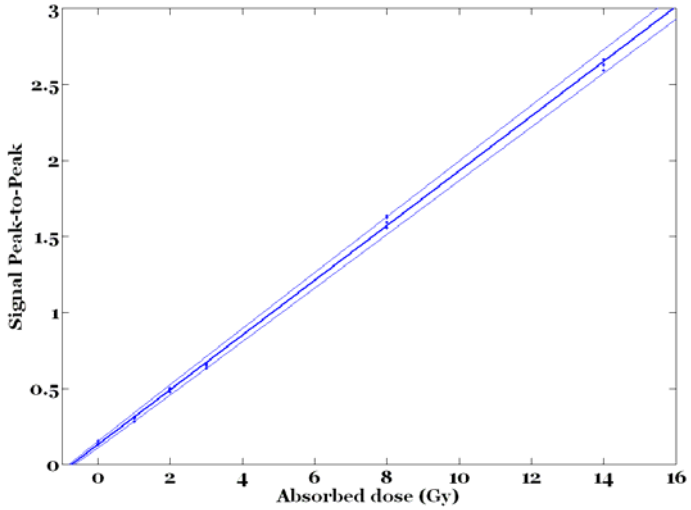

Figure 2. *Dose response curve for sucrose obtained from peak-to-peak values according to figure 1. The thinner lines indicate the limits of 2 standard deviations. The slope is calculated to β = 0.180 ± 0.002 (1 SD) and the minimum detectable dose was calculated to 0.06 Gy (1 SD).*

A minimum detectable dose is deduced to be 0.12 Gy (2 SD) for sucrose. Nakajima (1988) reported 0.05 Gy as minimum detectable dose for sucrose. In figure 3 spectra of Stimorol irradiated to A = 0 Gy and B = 20 Gy are shown. No Mn^{2+}/MgO reference was used, but there are still strong signals at about 331 and 341 mT. These originate from the sample itself and indicate that there is manganese in the sample, probably originating from the gum base.

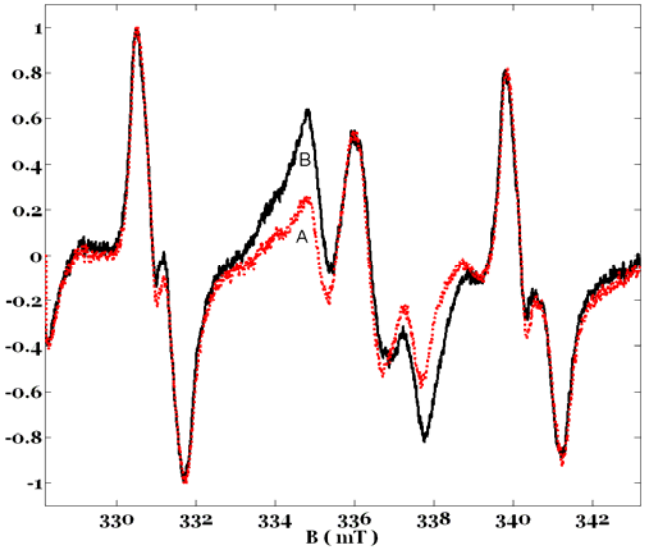

Figure 3. *EPR spectra from Stimorol. The spectra are normalized to the peak at 331 mT. The modulation amplitude was 0.5 mT. The given doses were A = 0 Gy and B = 20 Gy.*

V6 was chosen for the dose response measurements of chewing gum, since it yielded the smallest background signal of the four investigated chewing gums. Figure 4 shows the EPR spectra from V6, the background signal (A) and spectra from 5 (B) and 10 Gy (C). All spectra are normalized to the reference signals at 331 and 340 mT. The spectra show relatively large background signals at 336.5 mT. When acquiring the peak-to-peak values the maximum value was set to be measured at region I (334 - 336 mT) and the minimum at region II (337 - 338 mT) to avoid the influence of this signal.

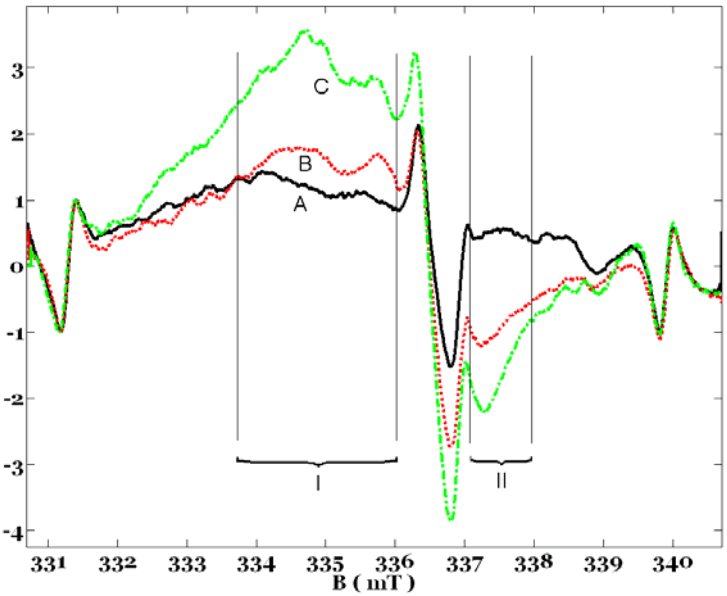

Figure 4. *EPR signal from V6 as a function of the applied magnetic field. The spectra are normalized to the Mn^{2+} reference peak at 331 mT. The modulation amplitude was 0.5 mT. "A" shows the background signal. The given doses were B = 5 Gy and C = 10 Gy. The peak-to-peak values are obtained from a maximum signal value inside region I and a minimum signal value inside region II.*

The corresponding dose response obtained 12 days after irradiation is shown in Figure 5.

The spectra were analyzed both after 7 and 12 days and the slope of the regression line increased from 0.35 to 0.42 during these days. This indicates an exchange among different radicals which is something that has to be further investigated.

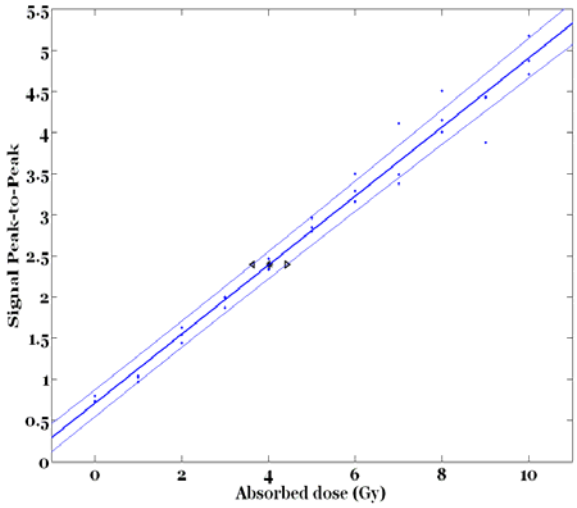

Figure 5. *Dose response curve for V6 obtained 12 days after irradiation. The thinner lines indicate the limits of 2 standard deviations. The slope is calculated to β = 0.42 ± 0.02 (1 SD). The dose of the blindtest is marked with a star and the corresponding error is marked with triangles and found to be 4.04 ± 0.40 Gy. The given dose was 3.96 Gy.*

The measured dose for the "unknown" sample was 4.04 ± 0.40 Gy (2 SD) for the measurement done 12 days after irradiation. The dose given to the "blind" tablet was 3.96 ± 0.02 Gy. Both measurements after 7 and 12 days respectively yielded a minimum detectable dose (2 SD) of 0.4 Gy.

Conclusion

These preliminary results show that chewing gums with sucrose as well as other sweetenings can be used for retrospective EPR dose determinations. The complex spectra probably originate from different substances and further experiments of irradiated sweetenings as xylitol, aspartam and sorbitol are in progress. The signal evolution with time needs to be further investigated. With improved knowledge from this together with the possibility to use a spectrometer with a 7 times higher sensitivity the accuracy will increase and doses below 100 mGy will be possible to determine with an acceptable accuracy.

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Trenger vi jodtabletter i Norge?

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Statens strålevern, Norge

Sammendrag

Anbefaling om inntak av jodtabletter er ett av flere tiltak som kan iverksettes ved en atomhendelse for å redusere doser fra inhalasjon av radioaktivt jod. Dagens ordning for jodprofylakse i Norge har eksistert siden 2002. I tillegg til et sentrallager i Oslo, har jodtabletter blitt distribuert til kommunene i de tre nordligste fylkene nord for området Salten med tanke på jodprofylakse rettet mot barn. Det har også vært gitt anbefaling fra Statens strålevern om å opprette lokale lagre i de kommunene hvor det er forskningsreaktorer eller anløpshavner for reaktordrevne fartøy. Det siste året har tiltaket vært oppe til ny vurdering i Norge. Tablettene bør fornyes i forhold til påstemplet holdbarhetstid, og dagens ordning har også skapt debatt i Nord-Norge. I arbeidet med å gjennomgå tiltaket har Statens strålevern samarbeidet med Helsedirektoratet. Aktuelle scenarier, mulige stråledoser, geografiske områder og aktuelle distribusjonsordninger har vært vurdert på nytt.

Innledning

"Utdeling av jodtabletter" er ett av flere dosereduserende tiltak som kan iverksettes av Kriseutvalget for atomberedskap ved en atomhendelse (Kgl. res. av 17. februar 2006, StrålevernHefte 29). Kriseutvalget vurderer nytten av tiltaket i Norge og gir anbefaling om innføring, gjennomføring og iverksettelse av tiltaket. Helsedirektoratet og Statens strålevern deler ansvaret for jodprofylakse, og begge direktoratene er medlemmer av Kriseutvalget. Helsedirektoratet har ansvar for bestilling, innkjøp og distribusjon av jodtabletter, mens Strålevernet vurderer det nasjonale behovet for tiltaket ut fra Strålevernets mandat om bl.a. å begrense stråledosene til befolkningen.

Dagens ordning med sentrallager i Oslo og distribusjon til kommunene i de tre nordligste fylkene nord for området Salten i Nordland fylke har eksistert siden 2002. I 2005 ble det gitt anbefaling fra Statens strålevern om å opprette lokale lagre i kommuner hvor det er forskningsreaktorer eller anløpshavner for reaktordrevne fartøy. På Kriseutvalgets møte i mars 2007 var det enighet om at tiltaket jodprofylakse burde opprettholdes. Det ble imidlertid besluttet at målgruppe for tiltaket, geografisk distribusjon og distribusjonsordning skulle vurderes på nytt. Kriseutvalget ba Statens strålevern utarbeide en faglig begrunnelse for jodprofylakse. I samarbeid med Helsedirektoratet har Statens strålevern det siste året arbeidet med en ny vurdering av tiltaket.

Historisk perspektiv

Jodprofylakse som et mulig tiltak i forbindelse med en potensiell atomulykke ble vurdert for første gang i Norge på midten av 1960-tallet. Dette skjedde i forbindelse med at den amerikanske atomdrevene passasjerbåten NS "Savannah" skulle ankomme Oslo og den tyske atomdrevne lastebåten "Otto Hahn" skulle transportere malm fra Kiruna langs norskekysten til Tyskland. På 1960-tallet ble det produsert 20 000 kaliumjodidtabletter, som stort sett ble lagret hos Statens Institutt for Strålehygiene (nå Statens Strålevern). En del tabletter ble også distribuert til havnene i Oslo, Stavanger og Bergen. Neste gang det ble anskaffet jodtabletter var i etterkant av Tsjernobylulykken. På slutten av

1980-tallet ble det distribuert tabletter til Nord-Norge, nord for området Salten i Nordland fylke. I tillegg var det et sentrallager hos Statens strålevern. Siden 2000 har Statens strålevern fulgt anbefalingene om tiltaksnivå og dosering av jod gitt i WHO-rapporten fra 1999 (WHO 1999). I 2002 ble det kjøpt inn nye jodtabletter.

Dagens ordning

Basert på gjennomgang av risikoen for jodutslipp til Norge, har fokuset for jodprofylakse tidligere først og fremst vært rettet mot Nord-Norge. Siden Norge ikke har kjernekraft, har sannsynligheten for utslipp av radioaktivt jod vært vurdert som lav og nesten utelukkende forbundet med utslipp fra utenlandske kjernekraftverk og reaktordrevne fartøy med fokus på nordområdene. I 1999 ble det utført en konsekvensanalyse av alvorlige scenarier for en potensiell ulykke ved Kola kjernekraftverk (StrålevernRapport 1999:10). I et av scenariene, hvor utslipp til Øst-Finnmark ble studert (Kirkenes-scenariet), ble mulig dose til skjoldbruskkjertelen vurdert til å være av samme størrelsesorden (barn 1-2 år: 10,6 mGy) som tiltaksnivået for barn anbefalt av WHO (10 mGy). På bakgrunn av bl.a. disse analysene ble det i 2002 besluttet å fortsatt distribuere tabletter til kommunene i Nordland nord for Salten, Troms og Finmark, og med tanke på jodprofylakse rettet mot barn.

I 2001-2002 ble det kjøpt inn ca. 2,2 millioner nye kaliumjodidtabletter til erstatning for de gamle. I tillegg til distribusjonen til Nord-Norge, ble det lagret ca. 2 millioner tabletter (pakninger à 10 tabletter kaliumjodid 65 mg) på et sentrallager i Oslo. Disse var tenkt at skulle brukes ved eventuelle uforutsigbare scenarier og/eller ved utslipp fra utenlandske atomdrevne fartøy langs norskekysten (Jaworska 2007).

Etter gjeldende regler kan reaktordrevne fartøy anløpe havner ved Tromsø, Bergen og Stavanger. Statens strålevern anbefalte i 2005 at det ble plassert ut jodtabletter i forbindelse med anløpshavnene i Stavanger og Bergen. Siden Tromsø ligger i Nord-Norge, var det allerede distribuert jodtabletter til kommunen.

Institutt for Energiteknikk (IFE) har sine egne lagre av jodtabletter ved forskningsreaktorene på Kjeller (2 MW_t) og i Halden (25 MW_t). Disse tablettene er først og fremst beregnet på ansatte hos IFE. I 2006 ble kommunene Skedsmo og Halden, hvor forskningsreaktorene ligger, gitt anbefalinger om å etablere beredskapslagre av jodtabletter.

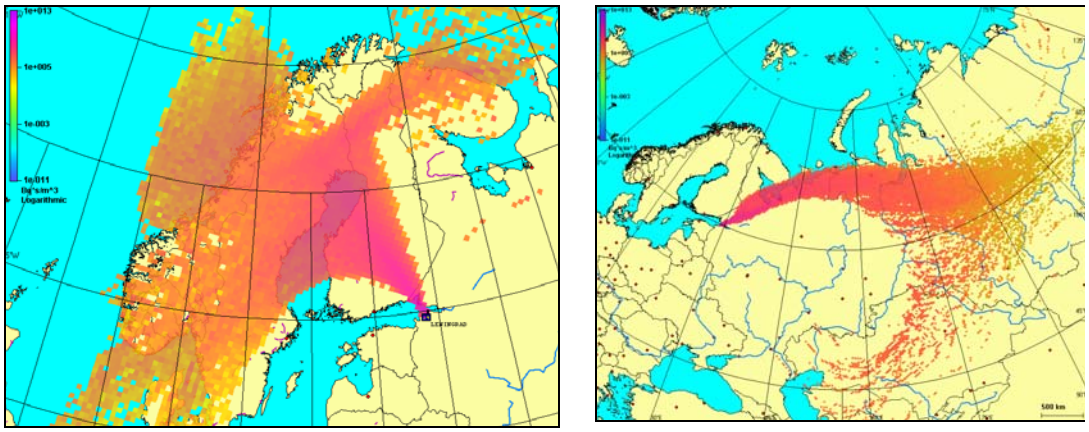
Ny vurdering av behovet for tiltaket

Ved vurderinger av strålingsnivåer og stråledoser er resultatet svært avhengig av hvilke forutsetninger som legges til grunn. Det vil være stor usikkerhet knyttet til vurderinger av stråledoser etter hypotetiske ulykker. Vi har likevel arbeidet med å samle eksisterende vurderinger av doser til skjoldbruskkjertelen og skaffe til veie noen nye data, for å etablere et bilde av hvilke stråledoser det kan dreie seg om ved ulike ulykkesscenarier. Anbefaling om jodtabletter vil vanligvis iverksettes i kombinasjon med innendørsopphold eller evakuering, noe som reduserer hvor stor dose som spares ved å ta jodtablett.

Ulykker ved kjernekraftverk og reaktordrevne fartøy i norske farvann vurderes som de største truslene med hensyn til utslipp av radioaktivt jod. De øst-europeiske kjernekraftverkene er av størst bekymring

når det gjelder ulykker ved kjernekraftverk. Kola kjernekraftverk ligger ca. 250 km fra den norske grensen, mens kjernekraftverkene Leningrad i nærheten av St. Petersburg i Russland og Ignalina i Litauen ligger henholdsvis ca. 900 og 960 km fra grensen til Norge. Sverige har kjernekraftverk omtrent 200 km fra den norske grensen. Det er også en rekke kjernekraftverk i størrelsesorden 500 km fra norske områder i Finland, Storbritannia og Tyskland.

Strålevernet har utført simuleringer av et potensielt utslipp fra en av reaktorene ved Leningrad kjernekraftverk med hensyn på radioaktivt jodnedfall og doser til skjoldbruskkjertelen. Simuleringene er utført i beslutningsstøttesystemet ARGOS (Accident Reporting and Guidance Operational system). Det ble brukt en langdistansemodell der det brukes reelle værdata og -prognoser fra Meteorologisk institutt, og det ble sett på flere ulike utslippstidspunkt. Siden vinden ofte har en annen retning enn mot Norge, ble det sett på størrelsen på nedfallet og doser til skjoldbruskkjertelen i en avstand 900 km fra reaktoren, tilsvarende avstanden til norske områder, uavhengig av vindretningen. Beregningene fra de fleste av simuleringene indikerte lave stråledoser til skjoldbruskkjertelen (0-1 mGy). I noen tilfeller var den beregnede joddosen til skjoldbruskkjertelen for barn (1 år) på samme nivå som WHOs tiltaksnivå for barn (10 mGy). For voksne var den beregnede dosen langt under WHOs tiltaksnivå for voksne (100 mGy).



Figur 1. *Simuleringer av utslipp etter en reaktorulykke ved Leningrad kraftverk. Figurene viser utslippenes trasé gjennom henholdsvis tre og fem døgn. Spredningsprognosene er hentet fra Meteorologisk institutt.*

Institutt for energiteknikk (IFE) har beregnet doser til skjoldbruskkjertelen ved uhell og påfølgende utslipp fra deres forskningsreaktorer på Kjeller og i Halden (IFE 2006a, IFE 2007b) i nærområdet i ulike avstander fra reaktorene. Særlig i nærområdet rundt reaktoren i Halden gir beregningene doser til skjoldbruskkjertelen i samme størrelsesorden som WHOs anbefalte tiltaksnivå for barn.

Det er få data knyttet til doser etter utslipp fra reaktorer brukt i fartøy. I 1992 vurderte en norsk offentlig utredning (NOU 1992:5) konsekvensene av reaktorhavari utenfor norskekysten. I utslippet 35 km fra land antas det at konsentrasjonen i luft av jod kan bli av en slik størrelse av det vil medføre stråledoser til skjoldbruskjertelen hos barn på 40-50 mGy.

Internasjonale aspekter

Jodprofylakse implementeres forskjellig i den nasjonale atomberedskapen i ulike land. Til og med europeiske naboland kan ha svært forskjellige anbefalinger. Det gjelder blant annet referansenivåer, aldersgrupper tiltaket anbefales for, størrelsen på sonen rundt kjernekraftverk hvor tablettene forhåndsdistribueres til husholdninger, distribusjonsordninger lokalt og tilgjengeligheten på apotek. I flere land er ordningen under revisjon. I land som ikke har kjernekraftverk etableres jodprofylakse stort sett på basis av trusselvurderingen av radioaktivt jodutslipp fra utenlandske installasjoner. Som eneste ikke-kjernekraftland forhåndsdistribuerte Irland i 2002 jodtabletter til alle husholdninger. Bakgrunnen for dette var en bekymring for en ulykke ved et utenlandsk kjernekraftverk, særlig på vestkysten av Storbritannia. Bekymringen ble forsterket etter terrorangrepene i USA i 2001, som viste at også utslipp fra kjernekraftverk som følge av et terrorangrep kan utgjøre en trussel. Det ble imidlertid nylig bestemt at ordningen ikke skal fornyes.

Norske anbefalinger for jodprofylakse baserer seg på WHOs anbefalinger fra 1999. Disse anbefalingene ble utviklet av WHOs regionskontor for Europa og er en revisjon av WHO-anbefalingene fra 1989. Rapporten fra 1999 anbefaler en avverget dose på 10 mGy til skjoldbruskkjertelen som tiltaksnivå for barn. For yngre voksne er tiltaksnivået 100 mGy.

Det er imidlertid en viss uenighet når det gjelder tiltaksnivåene og befolkningsgrupper tiltaket bør rette seg mot. IAEA anbefaler 100 mGy for alle aldersgrupper (IAEA Safety Series 115). IAEA Basic Safety Standards er under revisjon, og det antydes at det blir reduksjon i størrelsen på det anbefalte tiltaksnivået.

Det er enighet om at det er sterkt behov for internasjonal harmonisering av anbefalingene for jodprofylakse. Anbefalingene er under revidering i WHO, hvor IAEA er med i ekspertgruppen, og det arbeides nå for harmonisering av tiltaksnivåene i de nye anbefalingene fra de to FN-organisasjonene. Belgia, Frankrike, Tyskland, Luxembourg og Sveits har de siste par årene arbeidet med å harmonisere deres strategi i forhold til jodprofylakse og andre dosereduserende tiltak i de første timene etter en ulykke i et kjernekraftverk. Arbeidet til ekspertgruppen munnet ut i en rapport som ble gitt ut i 2007 (Ekspertgruppe fra fem europeiske naboland 2007). I 2006 ble det oppnådd konsensus blant de nordiske strålevernorganisasjonene om at man, i tråd med WHO-rapporten fra 1999, vil anbefale en avverget dose på 10 mGy til skjoldbruskkjertelen som tiltaksnivå for jodprofylakse til barn og unge.

Diskusjon

Den norske ordningen med å distribuere tabletter til kommuner er nokså unik. Det kan skyldes nærhet til russiske kjernekraftverk i nord, trafikk av reaktordrevne fartøy og de geografiske forholdene med blant annet store avstander. Den sørlige grensen for distribusjonen har fram til nå vært ved Salten i Nordland fylke. Grensen ble bestemt for flere titalls år siden i tråd med daværende risikovurdering. Det var da forventet anløp av reaktordrevne fartøy til havnen Bogen, litt nord for området Salten. I dagens situasjon er denne grensen vanskelig å begrunne. Ut fra en vurdering av mulige stråledoser ved ulike ulykkesscenarier, vil jodutslipp fra utenlandske kjernekraftverk og reaktordrevne fartøy også kunne gi inhalasjonsdoser av jod i andre deler av landet.

Det er rimelig at jodprofylakse videre planlegges mot utsatte grupper i forbindelse med anløpshavnene for reaktordrevne fartøy og Institutt for Energiteknikk sine forskningsreaktorer. I de øvrige delene av landet synes det å være grunnlag for å si at det bør være mulig å få tak i jodtabletter i rimelig tid til helt små barn, gravide og ammende kvinner, samt at det er informasjon om tiltaket rettet mot disse

befolkningsgruppene. En ordning med tilgjengelighet på apotek kan derfor være hensiktsmessig. De praktiske, økonomiske og administrative sidene ved tiltaket ligger innenfor Helsedirektoratets ansvarsområde. Disse aspektene utgjør en viktig del av beslutningsgrunnlaget for en fornyet ordning.

Opptaket av radioaktivt jod vil være avhengig av jodinnholdet i kosten, og vil være høyere dersom det er lite jod i kosten og tilsvarende lavere dersom kosten inneholder mye jod. Det har vært antatt at nivået på jod i norsk kosthold har vært høyt. Det finnes imidlertid ikke fullstendig data på dette området. Jodinnholdet i norsk kosthold ser ut til å være på tilstrekkelig nivå for de fleste, men avhenger sterkt av andelen meieriprodukter og fisk i kostholdet (Dahl et al 2003).

Påstemplet holdbarhetstid på de siste seriene av jodtabletter som er lagret sentralt og distribuert i Norge er fem år. Kaliumjodid er svært stabilt, og det er sannsynlig at tablettene er fullt brukbare langt utover formell holdbarhetstid. Til tross for at tester av kvaliteten på "gamle" jodtabletter har vist at de er fullt brukbare, er det likevel problematisk at tablettene "har gått ut på dato".

Det har vist seg å være et stort informasjonsbehov knyttet til tiltaket. Det er behov for tydelig informasjon mot flere forskjellige målgrupper (forvaltningen, helsevesenet, befolkningen). I en undersøkelse gjennomført i 2007 under Nasjonalt strålevernbarometer (Statens strålevern 2007), mener 70 % av befolkningen at det er sannsynlig at det skjer en ulykke ved et kjernekraftverk i Norges nærområder de neste 5-10 årene. Bare 15 % av de spurte hadde et godt inntrykk av myndighetenes innsats for å informere om forholdsregler ved en atomulykke. Dette understreker hvor viktig det er med god informasjon til befolkningen om atomberedskap og tiltak, herunder jodtabletter.

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7 Open session

Occupational radiation exposure from medical and industrial uses in Norway

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Abstract

In this paper, the dose statistics and dose distribution for $H_p[10]$ and $H_p[0.07]$ covering the years 2000 to 2007 from the national dosimetry service offered by the Norwegian Radiation Protection Authority (NRPA) are presented. Annual doses for different worker categories are presented together with the distribution of annual average doses in different dose intervals. Annual average doses for all monitored workers and for workers which received doses have also been determined.

The highest individual doses have been identified in medical uses of radiation, particularly in the medical professions radiologists and cardiologists. There has also been an increase in the annual doses for these groups over the past few years. In industrial uses of radiation no specific group of workers have been exposed to individual annual doses higher than previously registered. In this area of use, there has been no increase in the annual average doses.

The results show that the annual average doses for workers for which doses above 0.1 mSv has been reported has increased the last three to four years. This is mainly due to the contribution from the medical uses of radiation. The annual collective dose has decreased the last years, and is now approximately 2.5 manSv.

Introduction.

Norwegian employers are obligated by law and regulations to provide personal dosimetry for any worker that may be exposed to more than 1 mSv annually. Due to relatively low number of exposed workers in Norway, the Norwegian government has, as part of its services to the public, for 50 years offered personnel dosimetry services to all relevant industries and individuals. The Norwegian Radiation Protection Authority (NRPA) operates this national dosimetry service which presently includes the majority of occupational exposed workers in medical, industrial, research and educational uses of radiation in Norway.

Materials and methods

Approximately 7000 individuals are each year monitored by the dosimetry service, and the number of monitored workers is increasing. Monitoring periods are two months, and the operational quantities measured are $H_p[10]$ and $H_p[0.07]$ [1]. The dosimetry service is since 1999 based on thermoluminescence dosimetry. The NRPA dosimetry laboratory has two Harshaw Model 6600 TLD Readers and the dosimeters contain two element cards with TLD-100 from Harshaw. The minimum reported dose is 0.1 mSv. Dose results are reported to the employer with doses to individuals in the

present monitoring period and accumulated dose the present year. Background radiation corrections are made to the measurements. All reported doses are stored in a database at the NRPA. The monitored workers are categorised in different work categories (Figure 1). The categories have been established for covering the relevant work areas in Norway and for reporting the results to international organisations such as UNSCEAR. The dose statistics presented in this paper have thus been based on these categories and cover the period 2000-2007 [2, 3, 4]. The results from 2007 have not been published previously.

Results and discussion.

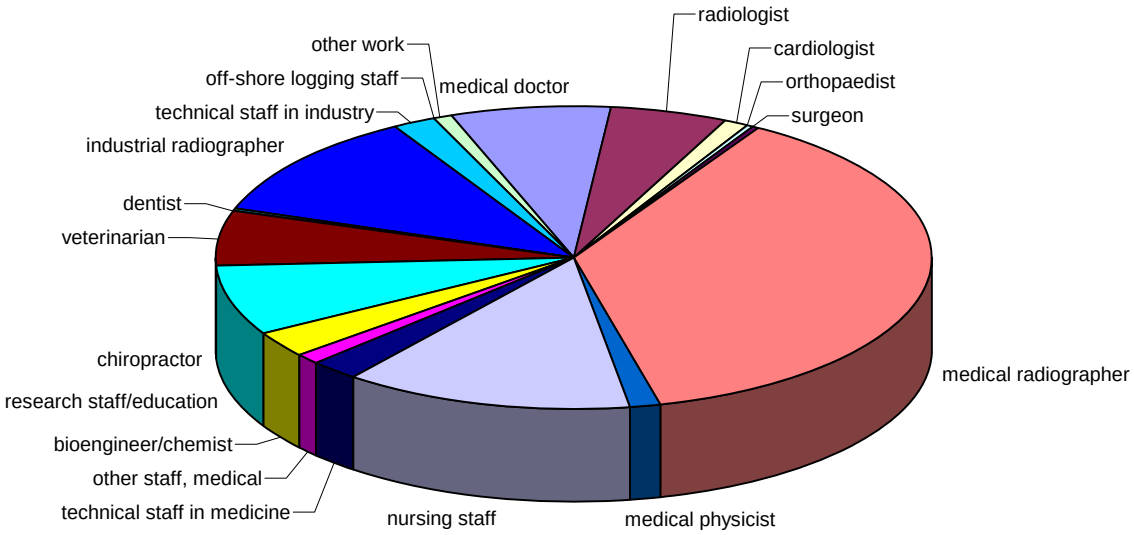

Figure 1. *Distribution of the different work categories based on the number of workers in each category. The figure is based on numbers from 2006.*

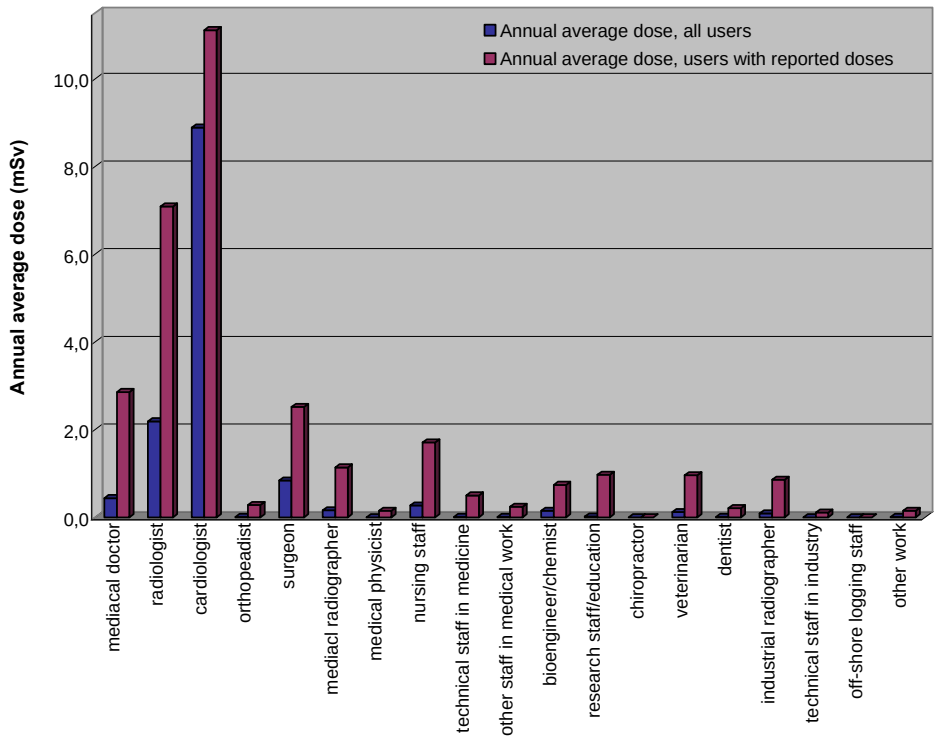

Figure 2. *Annual average doses for the different work categories. The figure is based on numbers from 2006.*

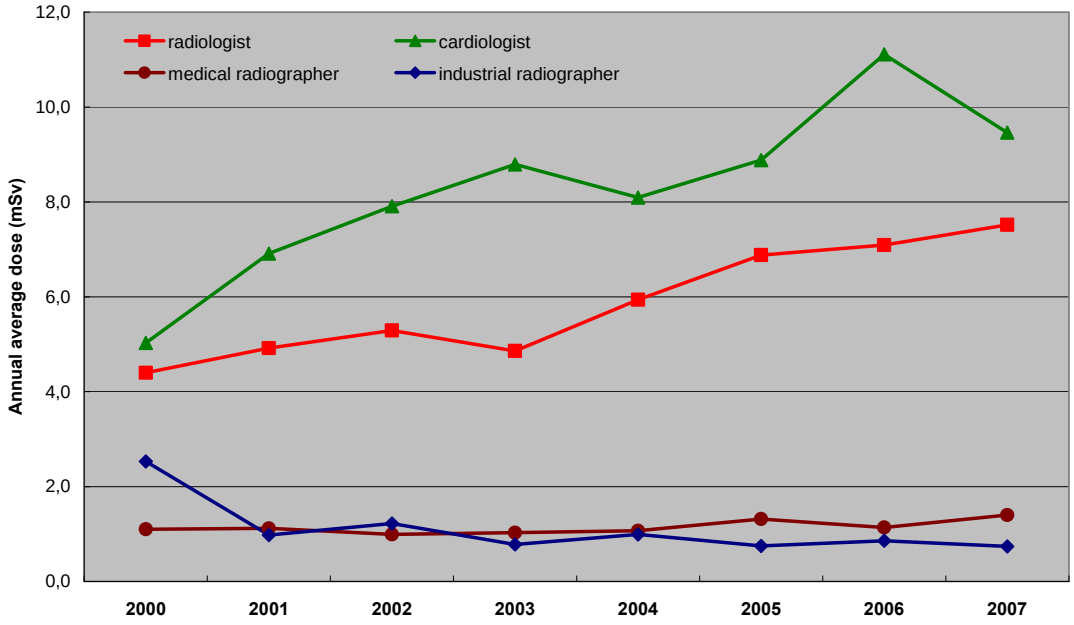

Figure 3. *Annual average doses for some selected work categories in the period 2000 to 2007.*

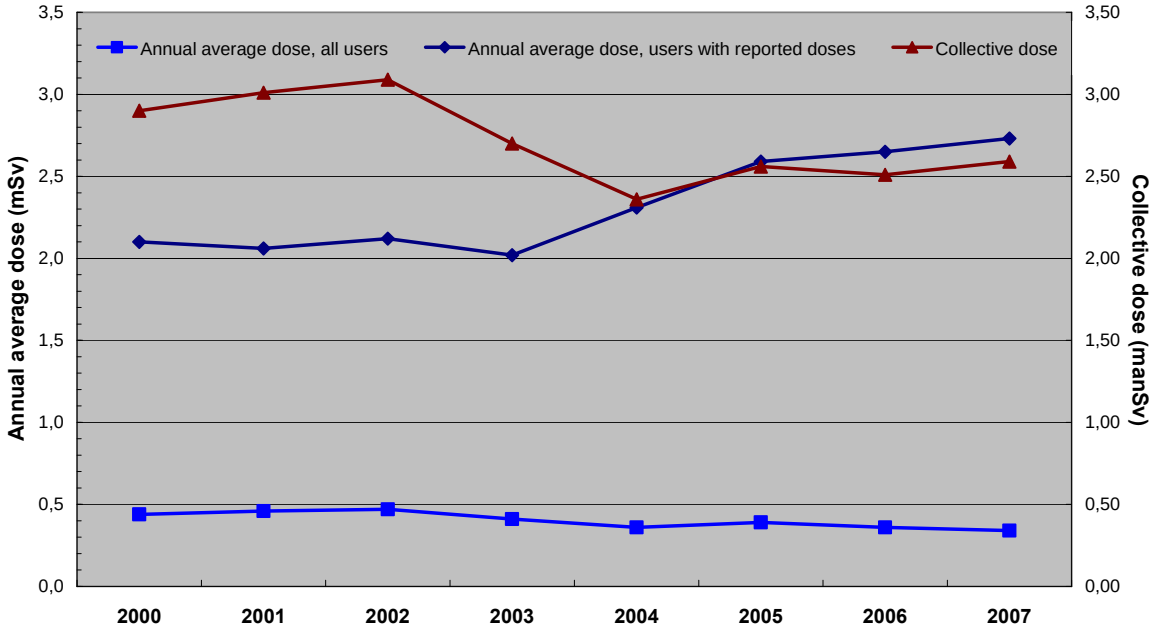

Figure 4. *Annual average doses and collective dose for all work categories in the period 2000 to 2007.*

In Figure 1 the distribution of the number of workers in the different work categories is presented. For 2006 the total number of workers was 6962 as seen in Table 1. Medical use of radiation has the largest group of workers. In 2006 medical use including veterinary medicine and dental radiation had 76 % of all users. The largest single group is medical radiographer with 38 % of all users. Industrial use of radiation has about 14 % of the users, the largest group being industrial radiography.

Annual average doses for 2006 for the different work categories are shown in Figure 2. Annual average doses for all dosimeter users was 0.36 mSv in 2006, and annual average doses for all users with doses above 0.1 mSv being the minimum reported dose for one measurement period was 2.65 mSv. The highest average annual doses are seen in the area of medical use covering several worker categories as seen in Figure 2. As shown in the figure two groups have considerably larger annual average doses than the rest of the dosimetry users. For cardiologists the annual average doses for users with doses above detection level is more than 10 mSv, and for radiologists above 7 mSv. Also the highest individual doses are found in these categories.

Figure 3 shows the annual average doses for individuals with reported doses for some selected work categories in the period 2000 to 2007. The annual average doses are approximately 1 mSv for industrial and medical radiography and have been at this level for the reported period. The annual average doses for the two groups, radiologist and cardiologist, have systematically increased the last years. No systematic analyses of the reason for this increase in these categories have been completed. It should be noted that if a protective lead apron have been in use, all measurements (see also Table 1) are obtained on the basis of having the dosimeter on the outside of the apron.

Figure 4 shows the annual average doses for personal dosimetry users at the NRPA dosimetry service for the reported period for all users and all users with reported doses (left y-axis), in addition collective

dose (right y-axis). The dose statistics for the years 2000 to 2007 is presented in Table 1. Figure 4 shows that the annual doses for users with detectable doses are increasing. At the same, as seen in Table 1, the number of users with doses reported is decreasing. A significant change with respect to the latter was registered in 2003-04 when the average number was reduced from approximately 1400 to about 1000 individuals. The reason for this shift has not been identified. The annual average doses for all users have been reduced in the reported period as seen in Table 1. The main reason for this is the large increase in individuals being measured without having any reported doses. The collective dose, which have been calculated on the basis of $H_p[10]$, has decreased and is now approximately 2.5 manSv per year.

Table 1.. *Dose statistics for the period 2000-2007.*

Year	No. of persons with annual average dose ($H_p[10]$, mSv)											Total	D=0		$\overline{D}$	$\overline{D}_{>0}$	KD
	[0-0,5>	[0,5-1>	[1-2>	[2-3>	[3-5>	[5-10>	[10-15>	[15-20>	[20-30>	[30-50>	50+		No.	%	mSv	mSv	manSv
2000	5862	221	196	103	104	67	27	16	9	1	3	6609	5225	79,06	0,44	2,1	2,9
2001	5699	274	214	95	92	80	25	16	7	6	4	6512	5050	77,55	0,46	2,06	3,01
2002	5744	219	244	84	95	68	22	14	15	7	3	6515	5058	77,64	0,47	2,12	3,09
2003	5801	242	202	80	73	71	26	14	10	3	3	6525	5188	79,51	0,41	2,02	2,7
2004	5949	199	140	74	69	56	23	10	9	3	3	6535	5511	84,33	0,36	2,31	2,36
2005	6039	174	152	60	63	69	29	11	8	5	3	6613	5628	85,1	0,39	2,59	2,56
2006	6412	157	136	72	68	54	29	15	8	9	2	6962	6016	86,41	0,36	2,65	2,51
2007	7065	165	121	71	74	51	23	12	20	4	3	7609	6663	87,57	0,34	2,73	2,59

Conclusions.

Approximately 7000 individuals are each year monitored by the dosimetry service offered by the Norwegian Radiation Protection Authority. For the period 2000 – 2007, all measurements ($H_p[10]$) have been assigned to worker categories covering most non-nuclear applications. On the basis of the registered doses, we have described the annual average doses for selected worker categories in the reported period in addition to the average annual doses for all categories for 2006. Also the collective dose has been calculated for the reported period. The results show that the collective dose has decreased from 2.9 to 2.6 manSv. However, the annual average doses have increased for users with doses above 0.1 mSv from 2.1 mSv to 2.7 mSv. Most significantly, the annual average doses for the medical worker categories radiologist and cardiologist have increased from 4.4 to 7.5 mSv and 5.0 to 9.5 mSv respectively.

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Event-Mode Data Acquisition for Non-Destructive Laboratory Analysis

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Abstract

Event-mode data acquisition and related software-based coincidence counting open new possibilities to sample analysis. STUK has launched a comprehensive R&D programme on non-destructive analysis (NDA) techniques that will address sampling, measurement and data analysis. A new measurement device, known as PANDA, is introduced. The device is expected to provide analysis results that in the past were only possible through time-consuming destructive analyses. PANDA will certainly improve detection limits and provide critical isotope ratios for nuclear forensics.

Introduction

High-resolution spectrometry is the most common way to determine the radionuclide contents of a sample. If a more complete picture on the sample properties is needed, results of various measurements have to be combined. However, only part of the information available is recorded. A single measurement is not able to respond to gamma, alpha and beta radiation.

To improve the quality of information received from the measurements, a research programme was started at STUK introducing up to date spectroscopy tools for sample analysis. This means, for example, introduction of position-sensitive detectors with a sophisticated data acquisition system. The device, known as PANDA (Particles And Non-Destructive Analysis), is under construction. The infrastructure of PANDA is finished and the device will be ready for test measurements at the end of summer 2008. The complete R&D programme, including development of new sampling and analysis techniques, lasts for several years; the current plans extend to the end of 2011.

The present paper discusses the use of $\alpha\beta\gamma Xe^{-}$ radiation for sample analysis and demonstrates progress in equipment development.

The following three items are addressed in the NDA programme of STUK [1]:

- **Sampling** - production of "matrix-free" samples;
- **Measurement** - capability to screen samples, to find individual particles of interest and to characterize them in detail; and
- **Analysis and data management** - capability to reveal the relevant information and to control the massive data volume.

Sampling

The quality of the sample is critical when no chemical processing is performed. To minimize the energy loss of the non-penetrating radiation, the active particles should be collected on the top of a smooth surface. The pile-up of individual particles is undesirable and the collection of large particles should be avoided.

Fibrous filters are widely used in air samplers (Fig 1). However, for NDA analyses, the membrane filters provide better quality because the particles are deposited on the surface of the filter. The sampling can be further improved by using methods based on the inertia of the particles.

The methods used in air sampling can be applied to collect a surface sample by vacuuming (Fig 2). When the particles are detached from the surfaces they behave like an aerosol and they can be collected on a membrane filter or on a plate.

Measurement

PANDA has two vacuum chambers, a loading chamber and a measurement chamber, that are separated by a gate valve (Fig. 3). PANDA is designed for simultaneous measurement of $\alpha\beta\gamma Xe^{-}$ radiation. The use of event-mode data acquisition system (also know as list mode) enables software-based coincidence and half-life studies [2].

The samples and the detectors are positioned against each other using two linear feedthroughs. The position accuracy is 10 μm. The sample can be moved between the measurement positions and also changed without venting the measurement chamber.

The measurement setup labelled "1" in Fig. 3 is primarily intended for screening of the samples and for the determination of their bulk properties. The setup includes a 64×64 mm^2 BB7 DSSSD from Micron Semiconductors and a planar germanium detector from Canberra with a crystal diameter of 70 mm and a thickness of 20 mm. The thickness of the DSSSD is 300 μm. This enables us to search for beta-emitting particles from the samples.

The compact measurement geometry of position 2 allows collimated views to a selected small area of the sample. Very high-resolution small spectrometers provide detailed information on the particle properties, such as isotope ratios.

Analysis and data management

STUK and its partners have designed a database, known as LINSSI (LINux System for Spectral Information) [3], for management of spectral data and related analyses. LINSSI is intended for the open-source platform (MySQL). Before PANDA's basic data, which is provided by the measurement position 1, can be utilized through the database, software needs to be written for converting the binary measurement data to a suitable XML format for database input. Special event-mode data handling capability will be designed for generating spectra with desired properties, such as an alpha spectrum from a certain pixel with the condition that the HPGe has detected 59.5 keV photons from ^{241}Am.

Discussion

PANDA is a unique device allowing particle-specific radionuclide determination that has not been possible in the past. This improvement is due to the software-based coincidence techniques. Feasibility studies show [2] that the spectrum baseline (background) is drastically reduced, even by a factor of 1000, as compared with traditional gamma spectrometric measurements. The signal is also weaker but only by a factor of 2. Therefore, the signal-to-noise ratio is much better, as well as the detection capability. And even more importantly, individual particles can be found and isotope ratios of interest can be determined. This is crucial information for nuclear forensics. The detection sensitivity is 10^{-12} g for plutonium [1].

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Figure 1. *Air samples in the past and today. The traditional way of processing air filters for counting is to press the sample to the smallest possible volume. However, the original sample is destroyed making it difficult to perform any other studies. The counting efficiency is reduced very little in a geometry with larger diameters (77 mm filter, 80 mm detector, BEGe5030). The filters are not destroyed and subsamples suit further studies, such as ordinary alpha spectrometry or measurement with PANDA.*

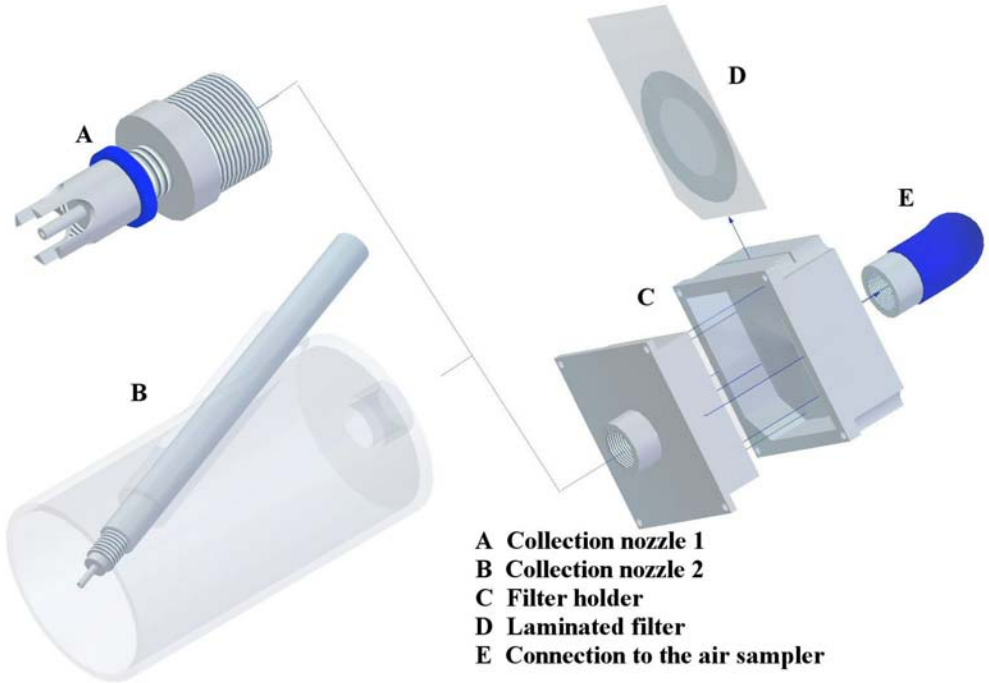

Figure 2. *Prototype devices for the collection of swipes using air flow. (A) Direct suction. (B) Shooting surface with a fast air stream; the air is accelerated in the nozzle and consequently, the particles are detached from the surface to a container; the small particles move together with the air flow whereas the large particles settle on the surfaces.*

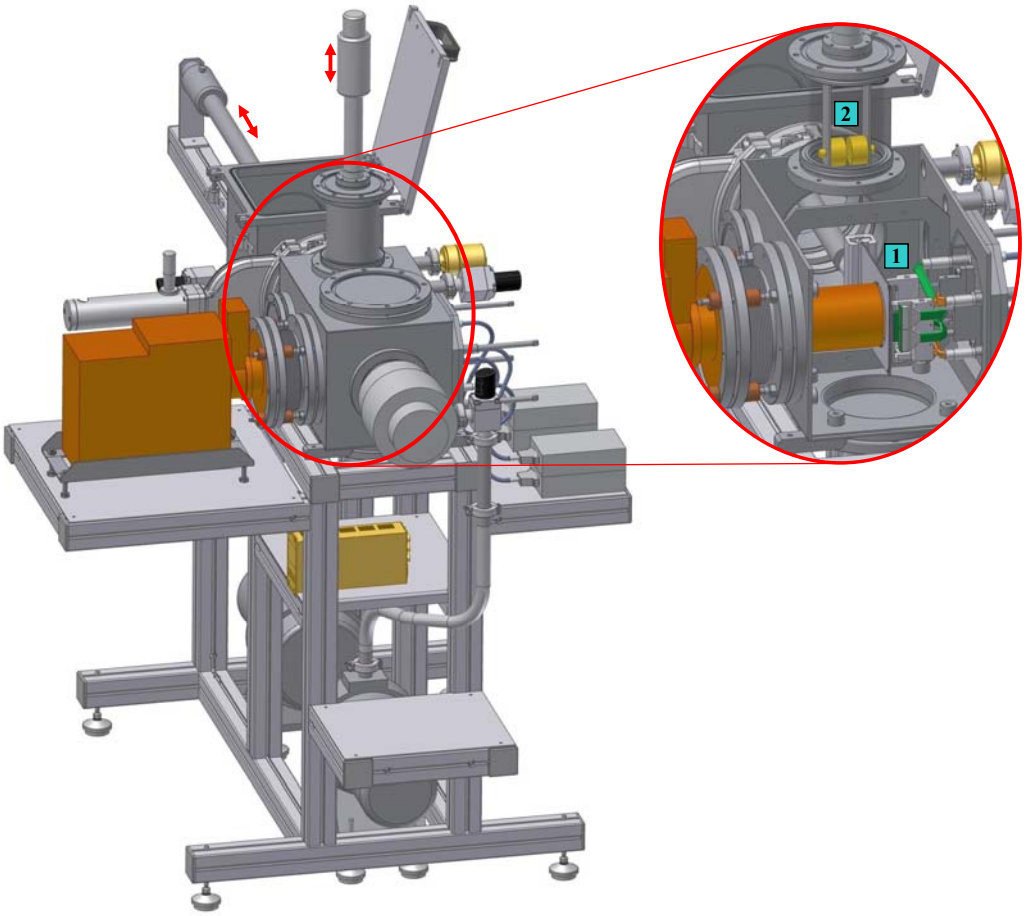

Figure 3. *PANDA. The device has a large vacuum chamber where different detectors can be mounted on measurement positions 1 and 2. In position 1, there is a large HPGe and a position-sensitive alpha spectrometer (32x32 pixels, each 2x2 mm^2). Position 2 is reserved for detailed particle characterization; the detectors are smaller than in position 1 but better in terms of resolution.*

The number and usage of sunbeds in Iceland 1988 and 2005

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Abstract

Reliable quantitative information on historic sunbed usage in Iceland is presently of great interest because it is a known risk factor for melanoma which incidence rate has increased rapidly, especially among women.

In this paper, data from two sunbed surveys in the years 1988 and 2005 are presented and discussed. Iceland has a relatively large number of sunbeds. In 1988 more than 1.5 sunbeds were listed per 1000 inhabitants living in the Reykjavik area. In the more recent survey from 2005, more than 1.0 sunbed was listed per 1000 inhabitants living outside the Reykjavik area. The data on sunbeds are supplemented by comparable Swedish data and information obtained in yearly telephone polls on sunbed usage, conducted since 2004. UVR exposure from sunbeds is estimated to be 2-3 tanning sessions per year, per person (all ages).

The data presented have been collected by the Icelandic Radiation Protection Institute in co-operation with the Environment and Food Agency, Capacent-Gallup, the Cancer society, Icelandic dermatologists and the Health directorate.

Introduction

Iceland used to have a lower rate of melanoma incidence than other Nordic countries which is as expected from its northern latitude, frequent cloud cover and consequently low natural UV-radiation. The Meteorological Office (www.vedur.is) reported 1268 sunhours per year in 1961-1990 which only gives 3.5 hours of sunshine, per day (and summer night) on average. In recent years it has measured the UV-index, with a maximum value of 4. - However, after a sharp increase in the incidence of melanoma that started in the 1980s, the melanoma incidence is no longer lower for Icelandic women (see *Figure 1*).

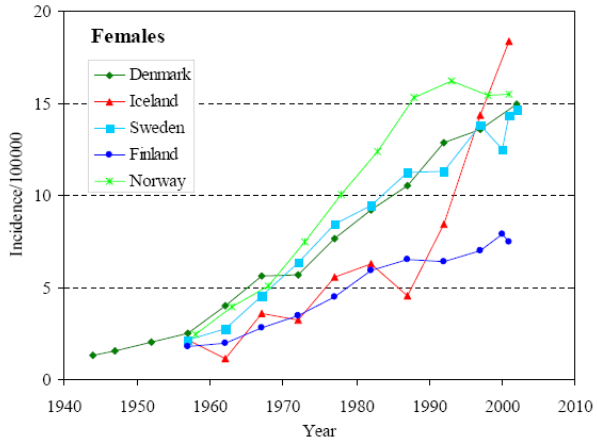

Figure 1. *The steadily increasing melanoma incidence per 100,000 in the Nordic countries among women, age standardized according to the WHO-world population, reproduced from reference [1].*

There are at least two possible explanations for the above. One is the introduction of modern sunbeds; the other is increase in travel to sunny destinations.

In this paper, previously unpublished data on sunbed numbers and usage is reviewed for the purpose of providing quantifiable information on UVR-doses received by the Icelandic population from

sunbeds for the last 20 years. These are shown to be large in comparison to doses in other countries. However, it will also be pointed out, with information on vacations in Spain in 1996 that doses received abroad may also have been significant.

Sunbed survey in 1988

Modern sunbeds were invented shortly before 1980. In 1979 there were only three sunbed saloons in Reykjavík[5] but their number increased rapidly. In a 1984 meeting[6] in Reykjavik, held to address concerns on sunbeds and skin cancer, there were 25 saloon[7] owners.

Table 1 *Number of services offering cosmetic tanning in Reykjavik.*

Reykjavík and neighborhood	1979	1984	1988	2005
No of saloons	3	25	56	28

In the summer of 1988, the Icelandic Radiation Protection Institute (IRPI or Geislavarnir) inspected every sunbed saloon in Reykjavik and some nearby municipalities (Seltjarnarnes, Kópavogur, Garðabær and Hafnarfjörður). The number of locations with sunbeds was 56, 25 of these were tanning saloons with 126 sunbeds, 20 were gyms and swimming halls and 11 were massage parlors, hairdressers and other services with tanning as secondary business[8]. A form was filled out for each sunbed with information on radiation intensity, spectral composition, markings, age of lamps, session times and more [2].

Table 2 *The IRPI survey in 1988. Population data is from Statistics Iceland (www.hagstofan.is).*

Sunbeds in 1988	No of Sunbeds	Sunbeds in tanning saloons	Population[9] 1st Dec 1988	Sunbeds per 1000 inhabitants
Reykjavik area	207	60.9 %	136,431	1.52
Rest of country	-		115,259	-

The average tanning session was 23.2 minutes with standard deviation 4 minutes (information available for 202 out of 207 sunbeds).

With information on how efficiently sunbeds are used, the number of tanning sessions can be calculated. We shall use information on Swedish sunbeds [3], [4] to estimate this figure. It is necessary to distinguish between sunbeds located in tanning saloons and sunbeds located elsewhere (in gyms, swimming halls, beauty parlors etc).

[5] Personal communication with owner of one of these saloons.
[6] According to a newspaper article in *Morgunblaðið*, page 50, 27. November 1984.
[7] It is not known if this number includes businesses with tanning as secondary business.
[8] The classification is based on the names of the businesses and is sometimes ambiguous.

Table 3 *Averages in tanning saloons compared with averages where tanning is not main business.*

	Session minutes in tanning saloons	No of beds in tanning saloons	Session minutes in sunbeds elsewhere	No of beds in services elsewhere	Ratio of session doses elsewhere to doses in tanning saloons
Reykjavik 1988	23.1	5.0	23.5	2.6	0.84
Gothenburg 2001	23.6	5.5	21.8	3.1	0.81

Sunbeds in tanning saloons produce larger UVR-doses per tanning session than sunbeds in other places. For the Gothenburg sunbeds, the ratio in the last column of *Table 3* is calculated for 21 sunbeds in tanning saloons versus 38 sunbeds in gyms and swimming halls and 2 in other businesses, from [4] page 11. For the sunbeds in Reykjavik, it is for 124 beds in tanning saloons vs. 59 in gyms and swimming halls and 18 in various other places. Session doses had a standard deviation of some 26 % which gives the 95 %-confidence interval as 0.78-0.90. We note this difference in doses, yet in what follows; we treat all tanning sessions as equal.

We expect lamps in tanning saloons to be exchanged more frequently and to be in more daily use, than lamps elsewhere. In the Reykjavik survey, the average age of lamps in tanning saloons was 224 hours (for 114 sunbeds, standard deviation 152 hours). For other places the average was 315 hours (for 59 sunbeds, standard deviation 215 hours).

In Gothenburg, an average number of 2250 tanning sessions was reported for 60 sunbeds in 11 tanning saloons. With each session lasting 23.6 minutes, each sunbed was in average use of 885 hours per year or 10.1 % of the time. If sunbeds in tanning saloons in Reykjavik in 1988 were used with this same efficiency, their lamps have been replaced on average two times per year (2≈ 885 / (2 x 224)) which is reasonable.

Information in Gothenburg was also collected on 113 sunbeds in 36 places other than tanning saloons; the majority of these (as in Reykjavik 1988) were gyms and swimming halls. Those were reported in average use 450 hours per year (calculated from a table on page 6 in [4]) or 5.1 % of the time. Sunbeds in tanning saloons were thus used with twice the efficiency of sunbeds in other installations (2 ≈ 885/450).

Assuming that sunbeds in tanning saloon and sunbeds in other installations were used with the same efficiency in Reykjavik 1988 as in Gothenburg 2001, the number of tanning sessions becomes[10] 2.8 per person in Reykjavik and neighborhood in 1988.

In the 2001 survey for Gothenburg the average reported number of tanning sessions per sunbed was 1600. A total of 300 sunbeds were listed for a population of 471 000 (from Statistics Sweden, www.scb.se) which leads to[11] 1.0 tanning session per person, per year. In the Swedish report, reference [4] page 11, the number of tanning sessions is estimated to be 1.4 per year, for individuals of age 16-74 year old, assumed to be 350 000 in Gothenburg.

[10] 2.8 ≈ (126 x 885 x 60 / 23.1 + (207-126) x 450 x 60 / 23.5) / 136,431
[11] 1.0 ≈ 1600 x 300 /471 000 or 1.0 ≈ 300 x (0.33 x 0.101 + 0.66 x 0.051) x 365.25 x 24 x 60 /22/471000

Sunbed survey in 2005

In 2005, the Icelandic Radiation Protection Institute conducted another sunbed survey in co-operation with the Environment and food agency (www.ust.is) which co-ordinates the work of local health authorities who inspect and issue licenses to all sunbed-saloons.

Sunbeds were listed in 28 locations in the Reykjavik area (see *Table 1*). The total number of sunbed locations in all of Iceland was 87. Tanning saloons were 31 of these.

Table 4 *The IRPI and UST-survey 2005.*

Sunbeds in 2005	No of Sunbeds	Sunbeds in tanning saloons	Population 1st Dec 2005	Sunbeds per 1000 inhabitants
Reykjavik area	144	85.4 %	177,603	0.81
Rest of country	133	38.3 %	121,801	1.09
Total	277	62.8 %	299,404	0.93

Sunbeds listed in Reykjavik in 2005 were almost exclusively in tanning saloons (see *Table 4*). There are relatively more sunbeds on the countryside, but less than half of these are in tanning saloons and assuming, as we did in the previous section, that tanning saloons use their sunbeds with twice the efficiency of other places (10 % vs. 5 %), we find little difference in the estimated number of tanning sessions per person (2 % more on the countryside).

For the whole country, 63 % of the sunbeds (174 of 277) were in tanning saloons. Using the same efficiency figures as earlier, we estimate the number of tanning sessions to be[12] 1.7 per person per year. Here we have assumed that the average sunbed session is still 23.2 minutes which it surely is not. Sunbeds in 2005 were allowed more radiation of type UVB than in 1988 and could deliver more UVR-dose in shorter time. This manifests itself in shorter sunbed sessions.

In the 2005 survey, information on length of tanning sessions was not systematically recorded. It is however known that longer sessions than 20 minutes were now rare while 20 minutes were still the most common (as in 1988). Each tanning saloon typically had one or two so-called 'turbo-sunbeds' with shorter sessions, down to 10 minutes. A survey of session times in year 2007 of 44 sunbeds in 6 tanning saloons in nearby municipalities of Reykjavik, gave an average of 17 minutes with a standard deviation of 5 minutes. Sunbeds elsewhere may have had longer session times but hardly longer than 20 minutes.

Deducing that the average session time in 2005 was somewhere between 20 and 17 minutes, the estimated number of tanning sessions must be raised to somewhere between 2.0 and 2.4 per person, per year.

User surveys 2004 - 2007

A UV-task group was formed in 2004 by IRPI, the Cancer Society, the Association of Dermatologists and the Health Directorate. On behalf of the group, Capacent-Gallup was enlisted to monitor sunbed usage in Iceland [5].

[12] $1.7 \approx (0.63 \times 0.101 + 0.37 \times 0.051) \times 365 \times 24 \times 60 / 23.2 \times 0.93 / 1000$

In March each year 2004-2007, a group of 1350 people, aged 16-75 year old, was randomly selected from the national registry and contacted by telephone. Responses were around 60 %, 857 out of 3251 said they used sunbed the previous 12 months, or 26.4 %. The ratio of female to male users was 1.6.

In *Table 5* is information on where 835 sunbed users lived. No difference is seen in the proportion of sunbed users in Reykjavik and elsewhere even if there are relatively more sunbeds on the countryside according to *Table 4*. This supports our previous assignment of double efficiency to sunbeds in tanning saloons, most of which are in Reykjavik.

Table 5 *Telephone polls, conducted in spring 2004-2007 among 16-75 year old.*

Telephone polls 2004-2007 Age 16-75	**Responders**	**Sunbed users**	**Percents**
Reykjavik and nearby municipalities	1818	474	26.1 %
Rest of country	1354	361	26.7 %
Total	3172	835	26.3 %

Information was collected from 857 sunbed users on how often they visited sunbeds the previous 12 months (*see Table 6*).

Table 6 *Frequency of sunbed visits by users in 2004-2007.*

2004-2007	**1-2 times per year**	**3-5 times per year**	**6-11 times per year**	**1-3 times per month**	**Once a week or more often**
Percentage in group	33.4 %	26.8 %	20.9 %	13.9 %	5.0 %

A simple way of calculating an average from *Table 6* is to assume the following averages for each group: 1.5, 4.0, 8.5, 24 and 78. The average number of visits becomes 10.6. A large uncertainty in this figure comes from assigning a number between 1 and 7 to the frequency of visits of the last group. We have used 1.5 visits per week, but we could also have used 2 which would have given 11.9 as an average.

The 10.6 visits per year for the 26.3 % of *Table 5* give 2.8 tanning sessions per year for those who are 16-75 year old. According to Statistics Iceland, these age-groups constitute about 71% of the whole population and thus we get 2.0 tanning sessions per year, per person in the whole population.

Summary and conclusions

The session numbers in *Table 7* for Iceland are estimated by assuming that sunbeds in Reykjavik 1988 and in Iceland 2005 were used with the same efficiency as in Gothenburg 2001 adjusted for a different proportion of sunbeds in tanning saloons. The estimated number of tanning sessions in Iceland in 2005 gets support from user surveys in 2004-2007. Those surveys can also be compared to two postal

surveys made by the Swedish Radiation Protection Institute in 2005 and 2006, each with 2000 participants [6] (see *Table 8*).

Table 7 *Number of sunbeds in Gothenburg and Iceland and estimated number of tanning sessions*

	Year	Sun beds no	Popul-ation 1st Dec	Beds per person x 1000	Beds in tanning saloons	Average tanning minutes	Sessions per person & per year
Reykjavik	1988	207	136,000	1.5	61 %	23.2	2.8
Gothenburg	2001	300	471,000	0.6	33 %	22.0	1.0
Iceland	2005	277	298,000	0.9	63 %	17-20	2.0-2.4

Table 8 *Sunbed user surveys in Sweden and Iceland.*

	Years	Size of survey	Age of group	Ratio of female to male users	Sessions[13] per adult & per year	Percent using sunbeds previous 12 months
Iceland	2004 – 2007	5400	16-75	1.6	2.8	26 %
Sweden	2005 – 2006	4000	18-74	2.1	1.2	15 %

The surveys use somewhat different year-classes, different mode of contact (telephone vs. post) at different time of year (spring vs. autumn). The difference in sunbed usage is however so great, both in these surveys and when comparing number of sunbeds in Gothenburg 2001 with those in Iceland, that it seems safe to conclude that Icelanders in this century have received more UVR-doses in sunbeds than the Swedes, the amount may be twofold.

In the United Kingdom, Diffey [7] uses a survey from 1996 with 6000 participants to estimate the number of tanning sessions, according to which 7 % of the UK population use sunbeds on the average 11 times per year. This amounts to 0.77 tanning sessions per person, per year which is only about one third of the number of sessions we have estimated in Iceland in 1988 – 2005.

Natural UV-radiation is low in Iceland and sunbeds may thus contribute a relatively large part of the total UVR-dose. Exponential increase in traveling by Icelanders may however have added to their exposure to the sun.

In a 1998 publication [8] from Statistics Iceland on Icelandic Tourist Patterns in 1996*,* page 54*,* the number of overnight stays of Icelanders in other countries is given as 1,826,100. For a population of 268,927 (middle of year 1996), this amounts to 6.8 days per person. Half (48.6 %) of these days were spent in only three countries; Spain, Denmark and the USA. The stays in Spain can be assumed to be mostly solar-vacations for sunbathing and relaxation with each day possibly roughly equivalent to one tanning session. The Spanish days (overnight stays) are 421,700 and amount to 1.6 days per person. This observation indicates that both tanning abroad and tanning in sunbeds may have contributed significantly to the observed increase in the melanoma incidence in Iceland after 1990.

[13] These are sessions per person of the age indicated in the column on the left, i.e. 16-75 year for Iceland.

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8 Emergency preparedness poster presentation

5 POSTERS

Triage, monitoring and treatment of people after malevolent exposure to ionizing radiation - a handbook

Tua Rahola and Maarit Muikku[1], Carlos Rojas-Palma and Klaas van der Meer [2], Astrid Liland, Alicja Jaworska and Ane Jerstad[3], Phil Kruse[4], George Etherington[5], Maria del Rosario Pérez and Zhanat Carr[6], Genowefa Smagala[7]

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[2] SCK•CEN - Belgian Nuclear Research Centre (Belgium)

[3]NRPA - Norwegian Radiation Protection Authority (Norway)

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Abstract

Until recently European national emergency response plans concentrated on accidents at nuclear power plants. However the focus has now shifted to being prepared also for malevolent use of ionising radiation. The main objective of the Euratom 6th Framework Programme project TMT Handbook was to produce a practicable handbook for the effective and timely triage, monitoring and treatment of people exposed to radiation following a malevolent act. The handbook will give advice on how to prepare the response for such incidents and how to handle the situation starting at the scene of the incident through to the long term follow up.

Introduction

In the aftermath of the Chernobyl accident European national emergency response plans have been focused on accidents at nuclear power plants. Recently, the perception of the increased threat has shifted the focus to being prepared also for malevolent use of ionising radiation. The malevolent use is aimed at creating disruption and panic in the society. The casualties will most likely be members of the public. The radiation exposure can range from very low to substantial and it could be combined with conventional injuries. The level of emergency response activities varies widely from one European country to another. Medical hospitals receiving a significant number of victims might be overwhelmed. It might also happen that the national capacity for response would be overwhelmed and assistance asked for from neighbouring countries. A European harmonisation for handling such situations is needed.

The European Commission through the Euratom 6th Framework Programme is co-sponsoring the specific targeted research project TMT handbook. The main objective of this project is to produce a practicable handbook for the effective and timely triage, monitoring and treatment of people exposed to radiation following a malevolent act. The project started with the collection of already published material to identify useful practices and provide a basis on which to build a clear handbook that was consistent, as far as possible, with the current thoughts in this field. References to a few important guides and publications are given in this presentation. The handbook is developed for use by Emergency Planning and Response Organisations across Europe, and in hospitals where radiation victims might be treated. It contains both general information and detailed suggestions for actions to be taken at the scene of the incident and in the hospitals by specialized teams in radiation protection, monitoring and medical treatment. The handbook will give advice on how to prepare the response for such incidents and how to handle the situation starting at the scene of the incident through to the long term follow up. It also gives advice on how to deal with the dissemination of information to the public and the people directly affected. It will be a useful tool for training purposes.

Before publication of the final version the draft of the handbook will be presented to European national emergency response organizations. The organisations will be encouraged to test and evaluate the material through emergency response exercises and stakeholder meetings. This will help in harmonising the handbook for use in the European countries. The aim of this paper is to give an overview of the contents of the TMT handbook.

Structure and contents of the handbook

The Handbook content focuses on topics specifically related to the radiological triage, monitoring and treatment necessary to respond to a malevolent act, while ensuring the appropriate protection of responding personnel. The Handbook is not intended as a complete check list for first responders, but instructions are given for actions and it can be used in the training of such personnel. Nor is the Handbook intended to include exhaustive descriptions of medical treatment of victims as such, since hospital staff are already trained for this.

The handbook is divided into seven chapters, annexes and a glossary.

Chapter A introduces the handbook.

Chapter B gives a short explanation on how to use the Handbook.

Chapter C gives a short summary of possible malevolent scenarios. The chapter does not include detailed descriptions of scenarios in order not to facilitate malevolent acts by disaffected groups.

Chapter D gives general guidelines on public information and communication strategies.

Chapter E is dedicated to the best practice for triage and monitoring of people exposed to ionising radiation following a malevolent act.

Chapter F describes the best medical treatment and long-term follow-up of affected victims.

Chapter G is dedicated to international cooperation on early warning and assistance.

Chapters E and F give direct guidance for incidental and accidental situations. To help users some useful additional information is provided in the annexes at the end of the handbook. Relevant medical, public health and other terms are defined in the glossary, according to their use in the text.

The structure of the handbook is such that the direct instructions to be used by the incident responders are separated from more general information.

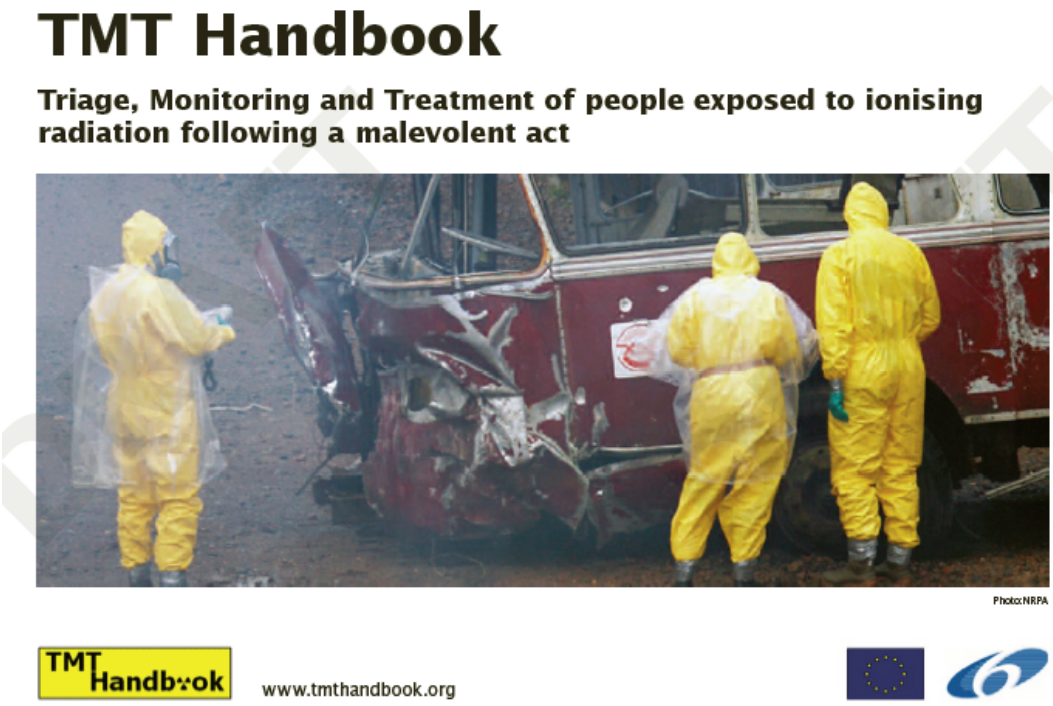

Figure 1. *Cover of the TMT Handbook draft*

Scenarios

In the beginning of the handbook a short summary of possible malevolent scenarios is given (Table 1). The chapter does not include detailed descriptions of scenarios in order not to facilitate malevolent acts by disaffected groups.

Table 1. *Some potential scenarios*

Threat to use radioactive material
Theft of radioactive material
Radiological Exposure Device
Radiological Dispersal Device
Attack on transport of radioactive material
Contamination of food and water supplies
Attack on nuclear installation or installation containing radioactive material
Improvised Nuclear Device

The scenarios are not meant to indicate the probability or possibility of any such event actually occurring. Neither should it be assumed that the scenarios described here are an exhaustive list of the possible incidents that could occur. It is believed that each country would have its own threat assessments as a basis for developing national radiation emergency and response plans. Some of these scenarios could result in received doses that might cause *Acute Radiation Syndrome (ARS)* in members of the public. For other scenarios, such exposures are unlikely. The two latter scenarios have not been elaborated further. An attack on a nuclear installation and subsequent release of radioactive material is well covered by the existing emergency plans, while an attack with an Improvised Nuclear Device was considered to be beyond the scope of the project.

Public information strategy

In the Handbook general guidelines on public information and communication strategies are described. It should be recognized that there exist cultural differences between countries and therefore similar means and techniques for communication may not be effective in all countries, any approach would need to be tailored to the specific situation and location. Public communication should be considered a key function in any response involving the malevolent use of radiation. Public communication activities can help prevent unnecessary fear or panic. Without successful public communication, authorities cannot achieve their emergency response objectives to save human lives and protect the public and the environment. Thus, information strategies constitute an integral part of emergency preparedness. For public communication to be credible and trustworthy, the organisation providing it must be seen as open and transparent. A plain language explanation of the radiation risks and any countermeasures being taken is a vital part of an effective risk communication process. It not only facilitates public understanding, it satisfies their need for information and fosters trust with those who are in charge.

Authorities, be it on a national, regional or local level, should prepare to respond to the public information demands that will arise during an event involving the malevolent use of radiation. To be effective, a public information response should be planned in advance of any emergency. These plans will need to be integrated within the overall planning for managing malevolent acts and should detail the roles and responsibilities to be carried out during the response.

Triage and monitoring

This chapter describes best practice for triage and monitoring of people exposed to ionising radiation following a malevolent act.

"Triage" is the use of simple procedures for rapidly sorting people into groups based on (a) their degree of physical injury and (b) on actual or potential effects on health, and the allocation of care to these people so as to expedite treatment and maximise the effective use of resources. The process is intended to maximise the number of survivors and can be termed "*trauma triage*". Trauma triage may be required following incidents involving the malevolent use of radiation or radioactive material in a public place. However, the scope of triage is broader for such incidents and includes a group of actions that can be termed "*radiological triage*". These actions are intended to sort people rapidly into groups depending on actual or potential effects on their health resulting from radiation exposure. In the handbook the objectives of the triage process are presented.

The term "*monitoring*" describes the measurement of radiation dose or contamination for reasons related to the assessment or control of exposure to radiation or radioactive material, and the interpretation of the results. The monitoring carried out in response to an incident involving the malevolent use of radiation or radioactive material in a public place may be subdivided into *source monitoring*, *environmental monitoring* and *individual monitoring*. The TMT Handbook is mainly concerned with individual monitoring, but the other forms of monitoring also come within the scope of the Handbook.

Source monitoring is the measurement of the activity of radioactive material or of external dose rates in the localised area around a source. In the present context, its main objectives are to locate and identify the source, and evaluate its potential for exposing people to radiation or radioactive material.

Environmental monitoring is the measurement of external dose rates in the environment, or of widespread contamination by radionuclides of environmental media such as air concentration or surface concentrations. In the present context, its main objective is to determine the geographic distributions of dose rates and/or levels of contamination.

Individual monitoring is monitoring using measurements of quantities of radioactive material in or on the body of the individual, or measurements made by equipment worn by individual workers. It includes the assessment of radiation doses to the individual from the results of such measurements. These measurements can be done with simple handheld instruments or more sophisticated ones like transportable monitors and whole-body counters.

Medical management at the hospital

The best medical treatment and long-term follow-up of affected victims are described. This chapter of the handbook is directed at doctors, nurses and other health workers who are responsible for actions to be taken at the first referral level (hospital response) concerning diagnosis, treatment and health care management of people affected by events involving the malevolent use of radioactive sources. It presents up-to-date guidelines for both inpatient and outpatient care. It was conceived to be used in hospitals where basic laboratory facilities and essential drugs and medicines are available. The guidelines presented are the result of a harmonized approach across the EU and are consistent with currently existing international guidance. Actions recommended are evidence-based statements systematically developed to assist decisions about appropriate health interventions. In areas where clinical evidence is limited, recommendations were based on expert criteria according with lessons identified in recent radiation incidents.

The chapter includes the management of ARS, local radiation injuries, radionuclide contamination and combined injuries. Emergency mental health care is also considered for prevention and management of psychological effects. Although the chapter is focused on clinical procedures, some public health actions are also described. Where appropriate, an individual (patient-based) approach is presented together with the public health (population-based) approach. Criteria for long-term surveillance of exposed individuals and populations are also presented.

International liaison

Nuclear and radiological accidents and situations resulting from malevolent acts involving radioactive material can become a serious threat to life, health, the environment and the society over wide geographical areas. The proper handling of serious nuclear and radiological emergencies, or in situations where prompt response is warranted in order to mitigate the effects of a perceived hazard, may require resources that challenge the capabilities of single Member States. It is therefore important for Member States to co-operate in order to better respond to such emergencies and situations through the arrangements set up in the international conventions on early notification and assistance.

The international conventions on early notification and assistance do not necessarily eliminate the need for the Member States to have additional bilateral or multilateral agreements relating to the information exchange or assistance. After the Chernobyl disaster in 1986, the number of bilateral or multilateral agreements has increased considerably.

Dissemination of the TMT Handbook

Before publication of the final version the draft of the handbook will be presented to European national emergency response organisations. The organisations will be encouraged to test and evaluate the material through emergency response exercises and stakeholder meetings. The testing period will be concluded at the feedback workshop in Norway in December 2008. The feedback will be the taken into consideration in developing the final handbook. A wide distribution is envisaged for the final TMT handbook and its incorporation into national exercise and training programs will be encouraged.

A TMT training course is planned to be held in February 2009. The training is directed primarily to national emergency response organisations with responsibility for first responders to emergency situations and the hospitals and wider health infrastructure, such as public health organisations. The aim of the course is to enable participants to enhance and strengthen capability and to contribute to the development of response systems and their coordination associated with the planning and response to acts involving the malevolent use of ionising radiation. In addition, the participants should be able to disseminate and promote the use of the TMT Handbook through the development and delivery of training tailored to the local situation in their home country. The course will provide a platform to identify common challenges and discuss opportunities for harmonised and coherent response strategies within the European Union.

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Nasjonalt overvåkningsnett for radioaktivitet i omgivelsene

Jan Erik Dyve og Bredo Møller

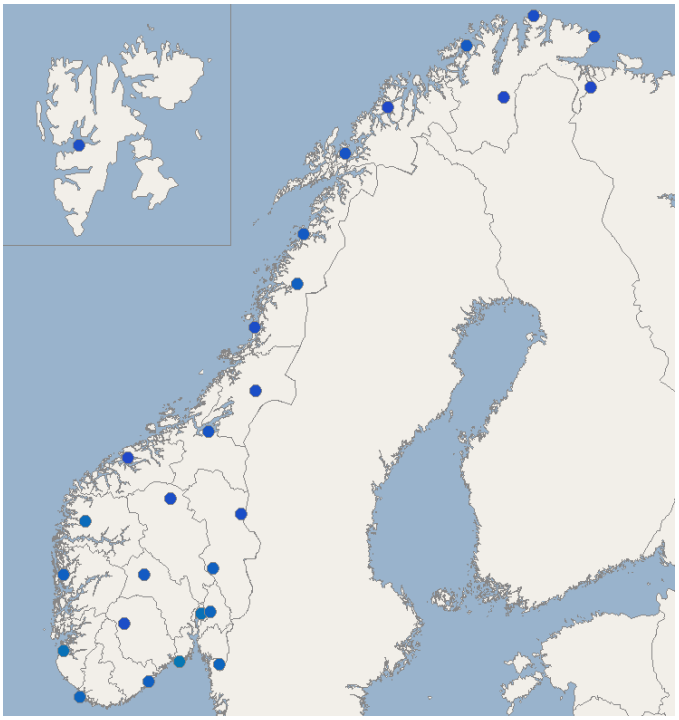
Statens strålevern

Abstract

Statens strålevern har ansvar for et landsdekkende nettverk av 28 stasjoner som kontinuerlig måler radioaktiviteten i omgivelsene. Nettverket ble etablert i årene etter Tsjernobyl-ulykken i 1986, og ble oppgradert til et nytt og moderne nettverk i 2006-2008. Formålet med målenettverket er å gi et tidlig varsel i tilfelle et ukjent radioaktivt utslipp rammer Norge. Videre vil målingene fra nettverket være en viktig del av beslutningsgrunnlaget i en tidlig fase når utslippet er kjent. Strålevernets telefonvakt mottar varsel på SMS dersom en stasjon måler forhøyede verdier.

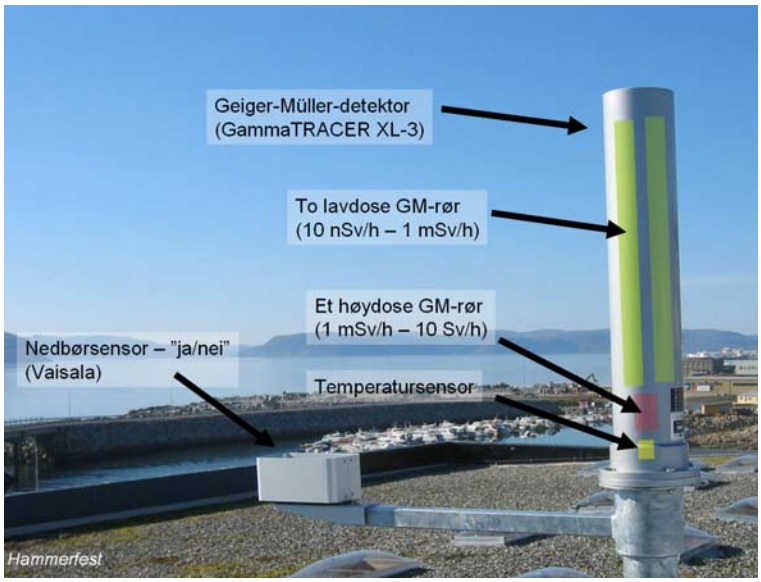
Plassering

Plasseringen av stasjonene er vurdert ut i fra forskjellige kriterier. God geografisk spredning, nærhet til befolkningstette områder og nærhet til nasjonale anlegg som anløpshavner og forskningsreaktorer. Dette innebærer minimum én stasjon i hvert fylke (inkludert en på Svalbard), men i for eksempel Finnmark er det plassert flere på grunn av nærheten til Nordvest-Russland. Stasjonene er som regel plassert på flat mark med avstand til hus og andre større objekter. I noen tilfeller er stasjonene plassert på tak eller vegg.



Oppbygging

En stasjon består av 2 utvendige detektorer og en datalogger plassert i et skap. Viktigste detektoren er GammaTracer XL-3 som måler doseraten i området 10 nSv/h-10 Sv/h. Den har tre GM-rør hvorav 2 er lavdoserør og siste er et høydoserør. De to lavdoserørene gir større volum og dermed bedre følsomhet i område under 1 mSv/h. Den andre detektoren er en nedbørssensor som registrere hvorvidt det er regn eller ikke. Dette gir verdifull informasjon ved en hendelse siden konsekvensene er langt større hvis det er nedbør. Nedbørsinformasjonen er også bra for å verifisere alarmer som skyldes radonutvasking.

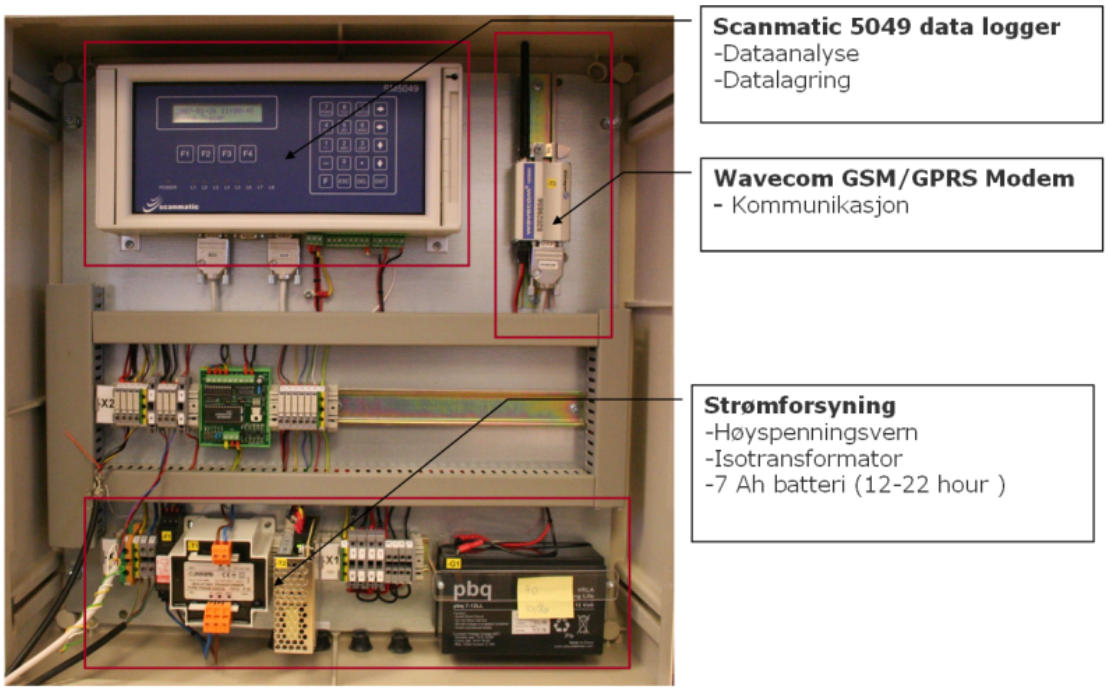


Figur 3: *Detektorene som sitter på Radnett-stasjonene.*

Begge detektorene er som regel plassert på en 3 meter høy mast. Høyden er bestemt ut fra en vurdering om mulig snø, fysisk sikring og mulighet for å plassere skapet på masta. Men i de fleste tilfeller er skapet plassert inn i en bygning i nærheten av stasjonen. På noen av stasjonene er detektorene plassert på en fellesmast (f.eks radioantenne) eller på bygning. Alle komponentene i stasjonen er valgt for å tåle nordisk klima. Figur 2 viser eksempler på installasjoner.

Figur 4*: Eksempler på forskjellige installasjoner.*

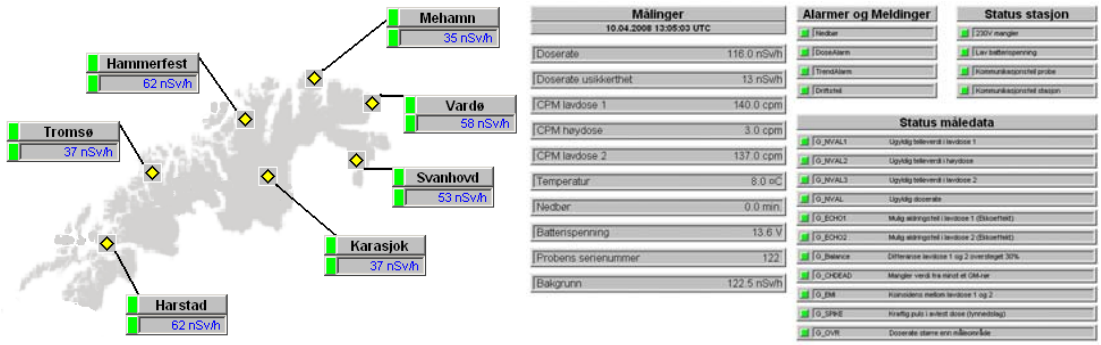
Skapet inneholder en datalogger som leser av verdier fra detektorene, analyserer og lagrer dataene. Loggeren kan brukes mot en rekke detektorer samt benytte flere kommunikasjonsveier. Den kan lagre inntil 8 uker med data. Skapet inneholder også strømforsyning, batteri og et GPRS-modem som sørger for kommunikasjon med innsamlingssystemet hos Statens strålevern. Figur nedenfor viser et bilde av komponentene i skapet.



Figur 5: *Komponentene i elektronikkskapet.*

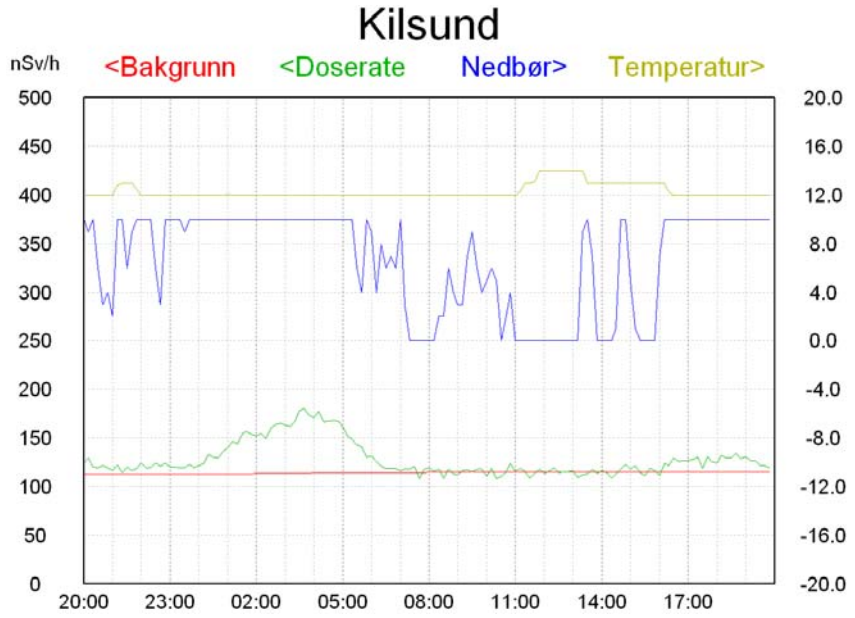
Databehandling

Stasjonene avleser detektorene hvert minutt og beregner et 10-minuttsnitt som rapporteres videre. For nedbørssensoren rapporteres summen, dvs. antall minutter med regn over denne perioden. Sammen med doseraten og nedbør rapporteres en rekke verdier som antall tellinger i GM-rørene, temperatur i detektoren, og en rekke kvalitetsparametere som sier noe om driftssituasjonen på stasjonen.



Figur 6: *IdacsView viser, til venstre, en oversikt over hele Radnett med informasjon om doserate, alarm- og driftsstatus. Videre kan den vise detaljert informasjon om målinger og status på målestasjonen (høyre).*

Hos Statens strålevern er det et innsamlingssystem som henter data fra alle stasjoner en gang i timen. Dataene blir lagret i en database og er tilgjenglig for vurdering og analyse gjennom egne verktøy, samt eksportert til beslutningsstøttesystemet ARGOS slik at det kan brukes sammen med andre måledata, modeller og spredningsprognoser i en krise. I det daglig brukes et verktøy som heter IdacsView. Dette gir en enkel oversikt over alle stasjoners gjeldende måleverdi og driftsstatus. Videre kan programmet vise detaljerte opplysninger om hver enkelt stasjon med muligheter for visning av trend og historisk data.



Figur 7: *Eksempel på visning av trender. Den økte doseraten skyldes radonutvasking. Merk hvordan toppen korresponderer med nedbør (blå linje).*

Varsling

Den viktigste oppgaven til Radnett er å vasle hvis en større, kjent eller ukjent, radioaktive forurensningssituasjon oppstår i Norge. Stasjonene analyserer data fortløpende gjennom å sammenlikne siste målte verdi med en bakgrunnsverdi som er basert snittet for den siste uken. Hvis differansen mellom måleverdien og bakgrunnen er større enn en forhåndsdefinert grenseverdi, går stasjonen i alarmmodus og sender et varsel til innsamlingssystemet. Innsamlingssystemet vil øke datainnhentingsfrekvensen til hvert 10 minutt og sende en SMS til telefonvakten. Denne må vurdere alarmen og evt. varsle videre til andre som kan gjøre en omfattende vurdering. Hvis en stasjon ikke har kontakt med innsamlingssystemet vil stasjonen, som reserverløsning, sende ut sms til definerte numre.

Datautveksling

Norge har datautvekslingsavtaler med landene i østersjøregionen og EU. Avtalen forplikter Strålevernet til hele tiden levere de siste dataene. Tilbake mottar Strålevernet tilsvarende data fra hele Europa. Totalt er det over 4000 målestasjoner som inngår i denne datautvekslingen. Datautvekslingen skjer kontinuerlig med minst mulig opphold, for å sikre andre aktører rask tilgang på data. I løpet av 2008 vil også publikum ha tilgang på måledata fra Radnett gjennom internett.

External radiation dose rate over the arctic ocean

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Abstract

Swedish Polar Research Secretariat arrenged a multidisciplinary scientific expedition to the Arctic Ocean in 2001 with the Swedish icebreaker Oden. The aim of the atmospheric research programme was to study the chemical, biological, physical and meteorological processes that control the formation of nanometre-size aerosol particles and their influence on climate change especially in the Arctic region. As a part of this programme external radiation was measured. The motivation was that ionising radiation has been suggested to be the reason for these particle formation events.

The expedition started from Gothenburg, Sweden, on 26 June 2001 and ended at Svalbard on 29 August. The expedition sailed first to Svalbard area, then NE to the Lomonosov ridge, the Makarov basin and to the North Pole. Most of the atmospheric programme was conducted during the ice drift experiment at the 88th latitude, while the icebreaker was moored to an ice floe and drifted for three weeks in August 2001.

Onboard the ship external radiation was measured with a pressurised ionisation chamber in 10 minute intervals. In the absence of gamma radiation coming from the Earth's crust the external radiation is practically due to the cosmic radiation. The absorbed dose rate varied between 0.040 and 0.045 μGy/h during the cruise. This corresponds to an ion production rate range of (7.3-8.3) x 10^9 ion pairs per hour per kilogramme of air. These values are low compared to continental regions. A linear correlation between the absorbed dose rate (DR) and barometric pressure (P) was found:

DR [μGy/h] = 0.124 – P [hPa] x 8.1E-05. This inverse dependance is due to the more efficient attenuation of the cosmic radiation in the atmosphere as the total mass of an air column increases.

Introduction

Swedish Polar Research Secretariat arrenged a multidisciplinary scientific expedition to the Arctic Ocean in 2001 with the Swedish icebreaker Oden. The aim of the atmospheric research programme was to study the chemical, biological, physical and meteorological processes that control the formation of nanometre-size aerosol particles and their influence on climate change especially in the Arctic region. As a part of this programme external radiation was measured. The motivation was that ionising radiation has been suggested to be the reason for these particle formation events.

Experimental

The Arctic Ocean 2001 expedition started from Gothenburg, Sweden, on 26 June 2001 and ended at Svalbard on 29 August. The expedition sailed first to Svalbard area, then NE to the Lomonosov ridge,

the Makarov basin and to the North Pole. Most of the atmospheric programme was conducted during the ice drift experiment at the 88th latitude, while the icebreaker was moored to an ice floe and drifted for three weeks in August 2001 (Fig. 1).

Onboard the ship external radiation was measured with a pressurised ionisation chamber (Eberline FHT191N) in 10 minute intervals. The instrument was calibrated at STUK – Radiation and nuclear safety authority, Finland. The effect of the ship on the dose rate was studied by temporarily measuring dose rate on the sea ice 500 m from the ship. The ship's radioactivity was found to increase the dose rate by 5 per cent and the observations onboard were corrected accordingly.

Results and discussion

In the absence of gamma radiation coming from the Earth's crust the external radiation is practically due to the cosmic radiation. The absorbed dose rate varied between 0.040 and 0.045 μGy/h during the cruise (Fig. 2). This corresponds to an ion production rate range of (7.3-8.3) x 10^9 ion pairs per hour per kilogramme of air. External radiation is the most important source of ionisation in the atmosphere as the amount of airborne radionuclides is extremely small (Paatero et al., 2004). These values are low compared to continental regions. In Finland the external dose rate varies usually between 0.03 and 0.30 μGy/h depending mainly on the natural radioactivity content of the surface soil and season. In winter the values are lower due to the attenuation of gamma radiation from the ground in the snow cover (Ristonmaa, 1998; Hatakka et al., 1998). In this study a linear correlation between the absorbed dose rate (DR) and barometric pressure (P) was found:

$$DR\ [\mu Gy/h] = 0.124 - P\ [hPa] \bullet\ 8.1 \bullet 10^{-5}. \quad (1)$$

This inverse dependance is due to the more efficient attenuation of the cosmic radiation in the atmosphere as the total mass of an air column increases (Fig. 3).

Acknowledgements

The authors would like to thank the Swedish Polar Research Secretariat for arrenging the expedition. The pleasant cooperation of captain, officers and crew of icebreaker Oden is gratefully acknowledged. This work was partially funded by Jenny and Antti Wihuri Foundation.

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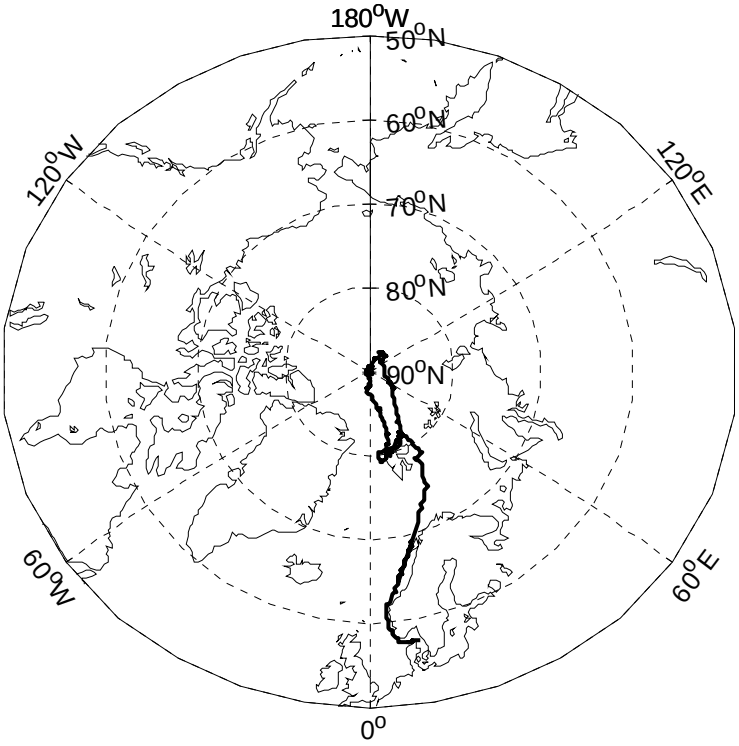

Figure 1. Expedition route.

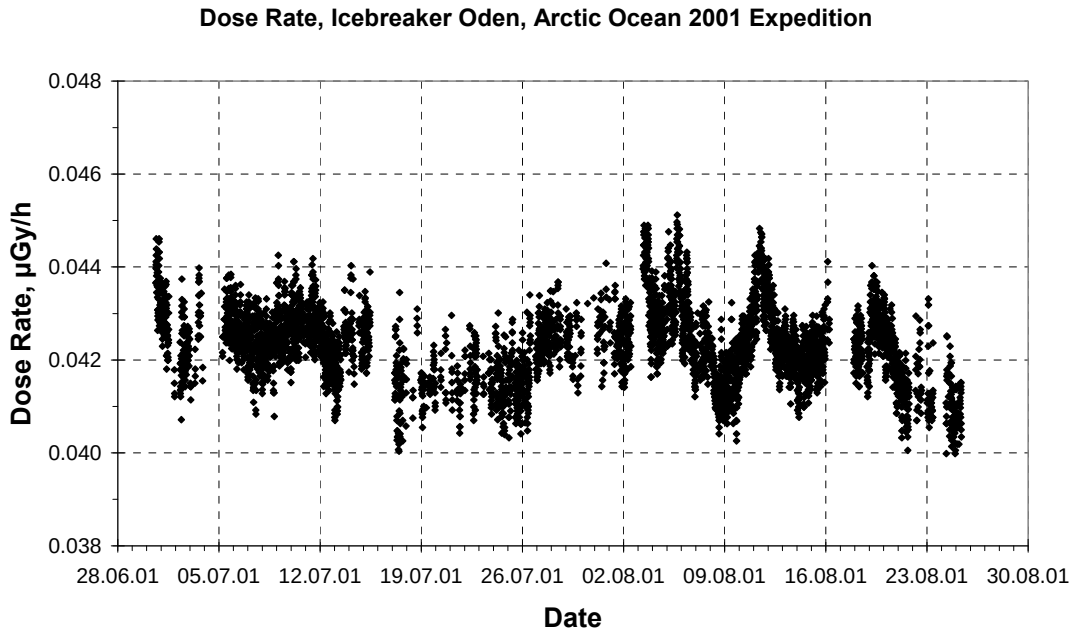

Figure 2. External radiation during the Arctic Ocean 2001 expedition.

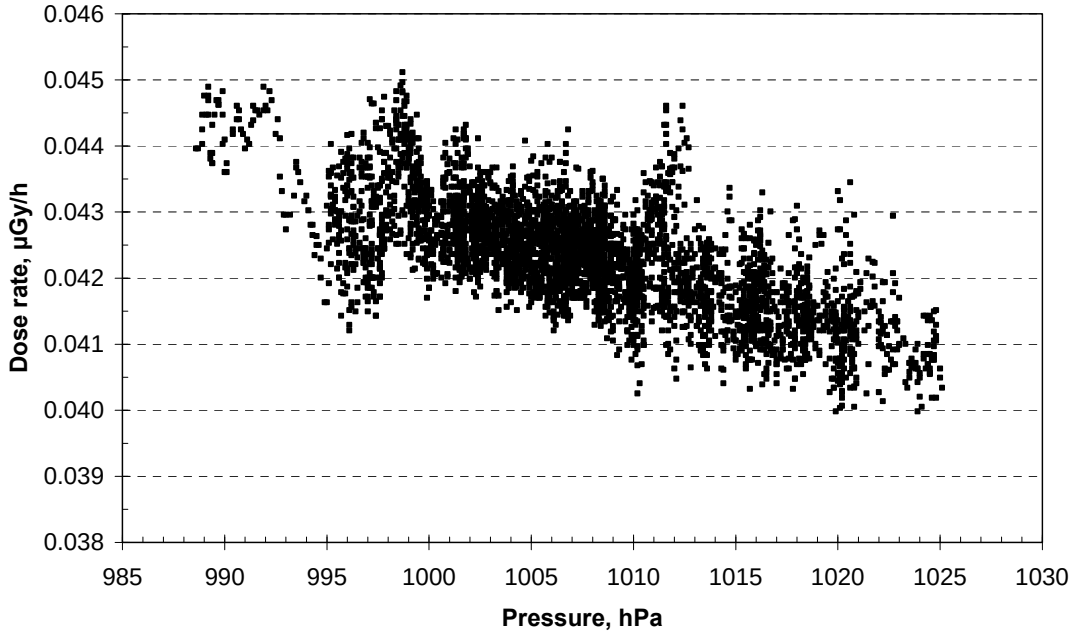

Figure 3. External radiation as a function of barometric pressure during the Arctic Ocean 2001 expedition.

A New mobile whole-body counter for measurement of internal contamination in Finland

Maarit Muikku, Jussi Huikari and Sauli Pusa

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Abstract

The internal radiation exposure of radiation workers and members of the public is determined using both *in-vivo* and bioassay measurements. STUK - Radiation and Nuclear Safety Authority has obtained a new mobile whole-body counter for measurements of internal contamination. The new mobile whole-body counter is used to assess the internal exposure of radiation workers and the exposure of the Finnish public. This unit can also be mobilised in emergency situations. The need for assessing internal radiation doses in emergency situations is evident which has been demonstrated after the accidents in e.g. Brasil and Ukraina.

Introduction

Assessment of internal radiation doses can be done using results from direct measurement of people or indirectly by excreta measurements. Estimations can also be made using air concentration data or activity concentrations in foodstuffs combined with consumption data. 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. *In-vivo* measurements are used to assess the internal exposure of radiation workers as well as the exposure of the public. 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. In situations with prolonged exposure repeated measurements are recommended. In emergency situations direct measurements should be done as soon as possible after an alert to give support for decision making and to reassure the general public.

Figure 1 *A new mobile whole-body counter.*

Method

STUK - Radiation and Nuclear Safety Authority has obtained a new mobile whole-body counter for measurements of internal contamination (Figure 1). The unit has been built on Volvo FE 42R truck which is powered by 6 cylinder 7.2 l and 206 kW turbodiesel. The torque available is 1050 Nm. The motor also is equipped with selective catalytic reduction technique so it already fulfils EURO V emission regulations. The power is transmitted by 6-speed gearbox and the smoothness of the ride is secured by air suspension, including automatic level control. The truck has non-blocking air powered disk breaks. The fuel consumption is about 30 l/100 km. The chassis is insulated and equipped with air conditioning, radiators (electric) and Webasto (diesel) heaters making all year usage comfortable. The monitoring unit uses 230 V AC, which is taken from the nearest wall. This external mains (230 V) is backed up by a Phoenix EasyPlus multifunctional energy system consisting of a powerful sinewave inverter, a sophisticated battery charger, a high speed AC transfer switch and AC distribution in a single light weight and compact enclosure. A battery pack (24 V/400 Ah) has been added in order to maintain measuring devices 24 hours in case of a loss of the external mains. This battery pack is also charged by the alternator of the truck when the engine is running.

The whole-body monitor inside the chassis consists of two HPGe detectors, digital electronics (Dspec Pro [1]) and a chair-shaped lead shield (Figure 2a). The detector set-up consists of a coaxial p-type HPGe-detector with a 90 % efficiency and so called GAMMA-X detector which is a coaxial n-type HPGe having an efficiency of 80 % (Figure 2b).The former detector is placed in the middle of the chair for whole-body measurements and the latter is placed closer to the upper body, providing the possibility to detect iodine accumulated in the thyroid, for example. The GAMMA-X has ultra thin entrance window made of beryllium, providing good efficiency also for low energy γ-rays. The detector end-caps are surrounded by a 5 cm thick lead shield. The typical time used in a routine measurement is 1000 s. Background is determined using the background phantom constructed from 5 kg sugar bags. If needed the measurement distance can be adjusted by lifting the detectors and the lead shield.

Figure 2 *Left: Inner view from the truck. Right: The two HPGe-detectors*

The efficiency calibration was performed using the adult St. Petersburg whole-body phantom [2] with ^{60}Co, ^{137}Cs, ^{40}K and ^{152}Eu rods (energy range 122-1460 keV, Figure 3). In addition, three St. Petersburg thyroid phantoms with ^{133}Ba capsules were used: adult, teenager (14 years old) and child (6 years old). The body burden of adult persons is determined roughly from knees to nose. The whole-body phantoms from 12 kg to 110 kg corresponding ages from two-year old to adult were used to obtain the correction to the efficiency due to the size of the measured person (Table 1). If the person to be measured is smaller than the adult F4 phantom used in the calibration, the body burden will be slightly overestimated.

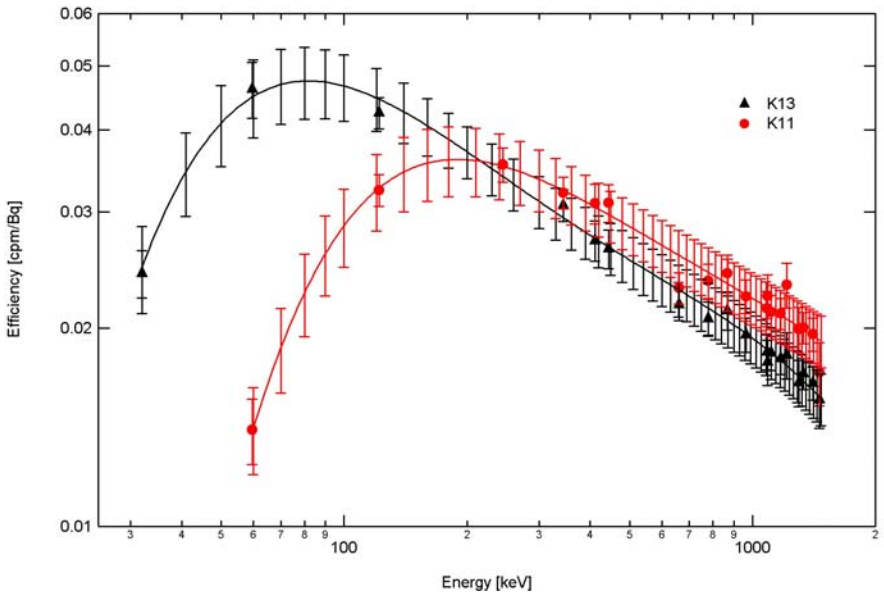

Figure 3 *Efficiency curves for the p-type HPGe (K11) and n-type GAMMA-X HPGe (K13) detectors.*

The uncertainty on the absolute γ-efficiency is 10 % for γ-rays >200 keV and goes up to 15 % towards lower energies. The final uncertainty on the activity measured will be determined by adding quadratically the statistical uncertainty of the identified γ-peak and that of the efficiency. For the most of the cases, uncertainties on the branching ratios and half-lives of the nuclei can be neglected. The minimum detectable activities (MDAs) [4] for the most commonly detected nuclei, ^{40}K, ^{137}Cs and ^{60}Co have been determined in two different locations (Table 2). The measurements were done in the garage in the first floor at STUK, Helsinki and in open air in commune of Padasjoki in Central Finland. As the MDA depends on the background level it should be determined when the background changes. The MDAs are higher in the garage due to the higher background from the ^{40}K in the surrounding concrete walls. In emergency situations the MDAs can be even higher due to the higher background from the environment.

Table 1. Correction to the efficiency due to the size of the measured person [3]

	Phantom			**Measured/Calculated**	
	Age [a]	**length [cm]**	**weight [kg]**	**K11**	**K13 (Gamma-X)**
F1	2	82.5	12	1.38 ± 0.07	1.10 ± 0.06
F2	6	121	24	1.21 ± 0.07	1.20 ± 0.07
F3	14	160	50	1.02 ± 0.06	0.96 ± 0.06
F4	≥18	170.5	70	1	1
F5	≥18	170.5	90	0.92 ± 0.05	0.93 ± 0.06
F6	≥18	170.5	110	1.00 ± 0.06	1.02 ± 0.06

In Finland the internal doses of radiation workers are registered, when the committed effective dose exceeds the registration limit of 0.1 mSv [5,6]. For example in the case of ^{60}Co (absorption class S), the level of 0.1 mSv is reached when the body content two weeks after an acute intake via inhalation is about 400 Bq. *In-vivo* monitoring can also be used to follow prolonged exposure e.g after a nuclear or radiological accident. If a chronic intake of ^{137}Cs via ingestion and constant body burden of about 3000 Bq is assumed, the annual effective dose is about 0.1 mSv. The accuracy of the new whole-body counter is good enough from the point of view of radiation protection. Most of the common radionuclides found in nuclear power plants, industry or radiomedicine can be determined at sufficient accuracy, if the normal measurement time of 1000s is used and if the radionuclides are homogenously distributed in human body. For screening purposes in emergency situations a shorter time can be chosen.

Table 2. *Minimum detectable activities (95% confidence) [3]. The detector K13 was not in use in Padasjoki.*

Garage, STUK Helsinki		**K11**	**K13 (Gamma-X)**
Nuclide	*Energy [keV]*	*MDA [Bq]*	*MDA [Bq]*
^{40}K	1460.8	3860	3790
^{60}Co	1173.2	120	130
”	1332.5	100	110
^{137}Cs	661.6	140	150
Padasjoki, Central Finland		**K11**	**K13 (Gamma-X)**
Nuclide	*Energy [keV]*	*MDA [Bq]*	*MDA [Bq]*
^{40}K	1460.8	1200	-
^{60}Co	1173.2	60	-
”	1332.5	60	-
^{137}Cs	661.6	80	-

Conclusions

In-vivo measurements are used to assess the internal exposure of radiation workers as well as the exposure of the public. In emergency situations it will be necessary to perform 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. The new mobile whole-body counter obtained by STUK - Radiation and Nuclear Safety Authority in Finland fulfils the requirements defined by the dose registration limits of radiation workers.

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Mobilt måleutstyr for måling av gammastråling

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Statens strålevern

Abstract

I 2006 fikk Statens stråleverns beredskapsenhet på Svanhovd i Sør-Varanger etablert utstyr for å gjennomføre mobile målinger. Utstyret kan brukes til å kartlegge radioaktiv forurensing over store områder på kort tid, til referansemålinger og til søk etter kilder på avveie. I 2007 er utstyret tatt aktivt i bruk bl.a. for å kartlegge store deler av Sør-Varanger og for å bistå Tollvesenet i Kirkenes ved Storskog med søk etter kilder eller annen forurensing. Utstyret representerer en utvidelse av målekapasiteten for norsk atomberedskap.

Ny kapasitet for norsk atomberedskap

I 2006 fikk Strålevernets beredskapsenhet på Svanhovd i Sør-Varanger etablert utstyr for å gjennomføre mobile gammaspektroskopiske målinger fra bil. Utstyret er svært effektivt og kan brukes til å kartlegge forurensing over store områder på kort tid. Høsten 2006 deltok beredskapsenheten med dette utstyret på en større øvelse i Sverige der målet var bl.a. å søke etter kilder på avveie. Utstyret viste seg å være effektivt og presist ved søk etter kilder. Sommeren 2007 ble det tatt aktivt i bruk bl.a. for å kartlegge strålenivået i Pasvik og ut til Grense Jakobselv ved den russiske grensen. I tillegg bistår Beredskapsenheten Tollvesenet i Kirkenes ved Storskog med søk etter kilder eller annen forurensing i miljøet.

Bil påmontert utstyr for deteksjon av radioaktive kilder under øvelsen DEMOEX i Sverige høsten 2006. Foto: Statens strålevern

I tillegg til bilmålinger er Strålevernet involvert i et samarbeid med Forsvaret for å ta i bruk tilsvarende utstyr i andre bærere. Det er planer om å bruke samme system i Forsvarets Orion-fly.

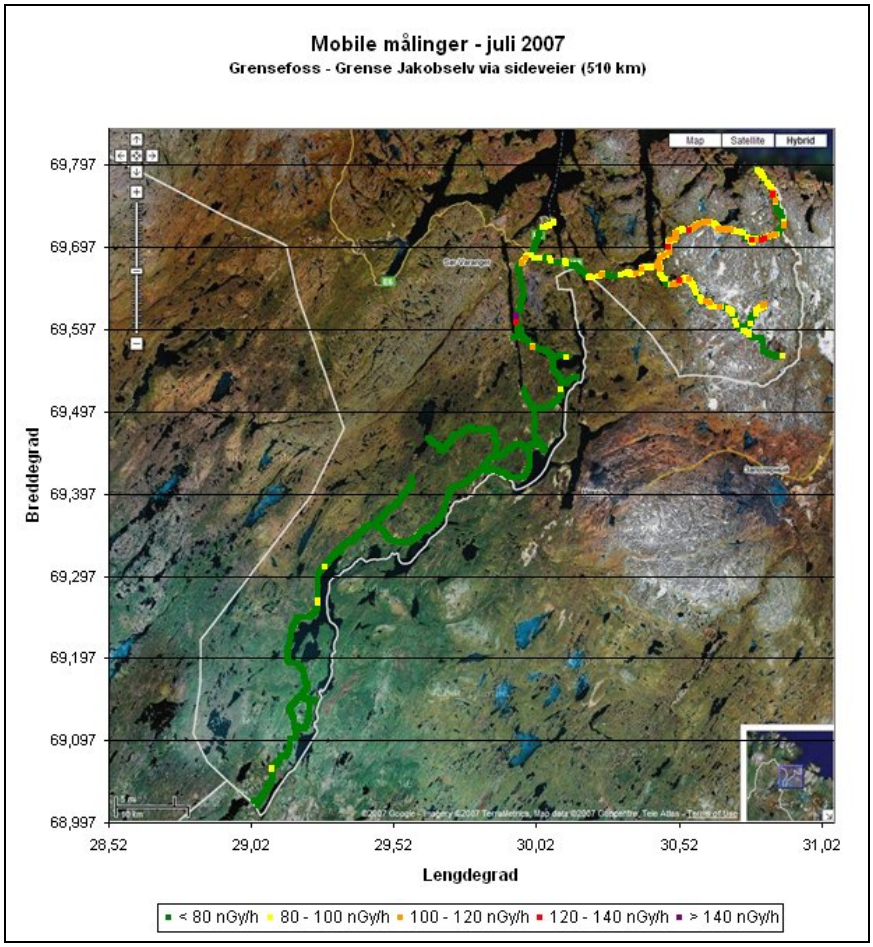
Det mobile måleutstyret består av 2 stk. 4 liter NaI-detektorer som ligger i hver sin skiboks som igjen er montert på hver side av bilens tak. NaI-detektorene måler gammastråling og er koblet til en elektronisk enhet (spektrometer) som behandler dataene som igjen sender signalene videre til en PC i bilen.

I tillegg er bilen utstyrt med bredbåndstilknytning via det digitaliserte NMT-nettet. Dette gjør overføring av data via internett til Strålevernets hovedkontor enkelt selv fra steder uten mobildekning. Dataene blir deretter presentert ved hjelp av egne beslutningsverktøy.

Resultater og presentasjon

Dataene som blir behandlet under kjøring lagres kontinuerlig og kan i ettertid tas ut og brukes i andre sammenhenger. For visning i sann tid kan PCen som er tilkoblet detektorene vise de mest relevante parametere og presentere dem grafisk på skjerm. Om man nærmer seg en kilde eller et punkt som har forhøyet strålenivå, vil dette trigge en alarm. Håndholdt utstyr, som også er tilgjengelig i bilen, brukes for å identifisere og kvantifisere eventuelle kilder.

Kartlegging av strålenivå i miljøet er viktig som referansemålinger. Med dagens trusselbilde kan man ikke utelukke et nytt radioaktivt nedfall over Norge. Ved å kartlegge nåværende strålenivå vil man etter et eventuelt nedfall vite mer om hvor mye miljøet er blitt forurenset sammenlignet med referansemålingene.



Referansemålinger i Pasvikdalen sommeren 2007 med over 50.000 målepunkter.

Utstyret vil også være til hjelp for Tollvesenet i Kirkenes. Med jevne mellomrom gjennomføres mobile målinger der Statens strålevern og Tollvesenet sammen "scanner" områder som ellers ikke vil være lett å kontrollere. Med dette utstyret har Tollvesenet nå mulighet for å få kontrollert skip liggende ved kai ved å kjøre langs denne. Dette vil naturligvis bare være et supplement til den normale kontrollen de utfører i forbindelse med inspeksjoner om bord på skipene.

Styrket atomberedskap

Utstyret for måling av gammastråling er enkelt å montere og kan gjøres operativt i løpet av kort tid (mindre enn 1 time). Måledata kan fremskaffes og resultater fra store områder videresendes, også fra områder uten mobildekning. Derfor er slikt måleutstyr en betydelig styrking av atomberedskapen i Norge.

9 Radioactive waste session

Principles for establishing criteria for disposal of all radioactive wastes

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1 Background

In countries with nuclear power, disposal of nuclear waste has been a priority during the last decades, in particular disposal of spent nuclear fuel. Regulation of non-nuclear waste has by comparison been lacking behind, but disposal of this type of waste has recently received more attention.

Guidance is now being produced from IAEA, which has started work to prepare a standard for disposal of all types of radioactive wastes, with the draft standard title "*DS354 Safety Requirement: Disposal of Radioactive Wastes"*. In the process, IAEA wants to cover all radioactive disposal by a single set of principles. Earlier, IAEA has covered near-surface and geological disposal by separate documents.

Below we present our understanding of the main principles for criteria-setting for all types of radioactive waste. Essentially, harmonization is achieved when due consideration are given to a number of factors, given in section 3. First, we mention some general policies and radiation protection policies in an introductory section.

2 Some general principles and complications

A general principle in all environmental protection work is the "polluter pays" principle, i.e. the one who pollutes should also take precautions, including providing the means, for limiting consequences for the environment.

In Sweden this principle is included in the decision to establish a nuclear waste fund, in connection with nuclear waste. The fund is established to ensure that means are available for all activities in connection with waste disposal and decommissioning, where radioactive waste is concerned. The principle is extended, by a recent European Commission Directive, to all High Activity Sealed Sources, HASS. In Sweden, no formal HASS fund is established, but a financial guarantee must be available to cover the disposal of the sources, and the requirement system is also expanded to cover essential all non-nuclear sources.

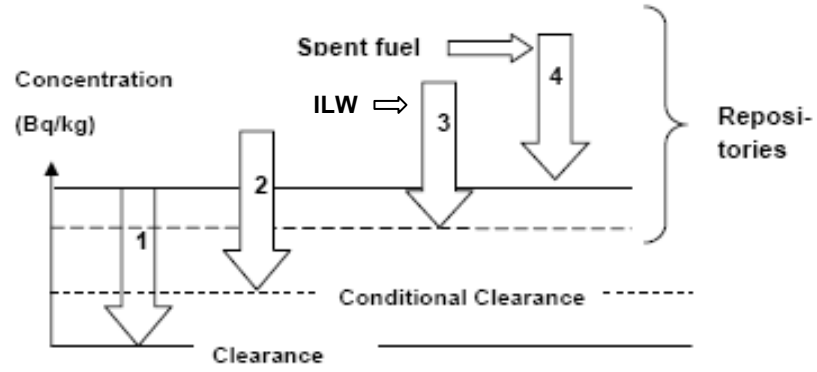
For NORM waste management, the picture is more complicated. A water filter attached to a water system from a private well may enrich uranium and other nuclides completely outside the "operator's" knowledge. The basics of radioactive waste management and disposal are less obvious in their application and the need for societal support in finding solutions is evident.

In the low end of the activity spectrum, exemption and clearance of sources are important radiation protection concepts. It is assumed that the reader is familiar with, e.g. the EURATOM Basic Safety Standards, BSS, and their lists of exemption values.

Radioactive waste can be disposed of with exemption from regulation if the waste falls under the appropriate clearance level for the waste stream in question. This level is sometimes derived from the exemptions levels, specified in the BSS. The principle of exemption in the BSS is justified by calculation of doses to the public and to workers from a number of chosen scenarios.

The scenario that yields the highest dose is called the limiting scenario. If the route of the waste is controlled in a conditional clearance process, so that fewer scenarios are relevant, making other scenarios limiting that have less severe consequences, disposal of waste with higher concentration may be accepted. Figure 1 shows 4 examples of disposal of waste with increasingly higher concentration: clearance (1), conditional clearance (2), disposal of Intermediate Level Waste ILW (3) and disposal of spent fuel (4).

Figure 1. Clearance and disposal at ~ 10(s of) µSv/a



A wide span of waste streams must be considered

The very idea of harmonization leads us to visit incongruent scenes of very high and very low level waste management, each associated with its own terminology and background.

On the high level waste side there is the activities of the so-called safety performance community, by i.a. OECD/NEA working groups and the European Commission's research projects, where the participants assemble at international meetings such as the Las Vegas International High Level Radioactive Waste Management Meeting, centred around the US Department of Energy's work on the Yucca Mountain repository.

At the other extreme we find the municipal waste disposal sites, landfills and other hosts of Very Low Level Radioactive Waste, VLLW, disposal. Requirements for disposal on these sites are related to rules for exemption and clearance. The system of harmonization factors in the next section were originally developed with licensed repositories in mind, but they may also be valuable for establishing criteria for conditional clearance for such waste streams as cesium-containing wood ash, waste for alum shale concrete and spent drinking water filters with radioactive substances. In the process of conditional clearance, the waste is subject to regulatory supervision until the conditions set up by the regulator – e.g. safe disposal in a municipal disposal site - are met. Another type of disposal situation

exists, whereby a municipal disposal site receives a license for disposal of radioactive waste. This direction of work is SSI's goal, which also enables the authority to have continued supervision.

3 Harmonization

A government investigation, on SSI's initiative, into non-nuclear radioactive waste has lead to a series of suggestions, i.a. clean-up funding for historic waste and orphan sources. A follow up investigation in this area formulates the authority's rules for disposal of non-nuclear radioactive wastes and to decide on disposal criteria for NORM waste on municipal disposal sites. As a first step in this direction SSI decided to formulate some general principles that may serve as guiding line for disposal of all types of radioactive waste.

SSI's goal is that radiation protection criteria used for disposal of all radioactive waste should be harmonized, taking all relevant circumstances into account. This can lead to different numerical values for different waste streams, and the same nuclide might be regulated differently, depending on the stream containing the material.

Conditions that may influence the rules for disposal has to do with a number of factors concerning the creation of the waste and other circumstances, that can be summed up in a number of societal, activity-related and technical factors given below. Harmonization is said to be achieved if these factors have been taken into account.

Societal factors

It is an obvious point of departure that activities related to waste management require resources and that policy- or political issues relating to economy and responsibility will play an important part, just as choices between alternatives practices creating waste.

In addition to such issues, mainly related to justification, societal and economical factors will also influence the boundary conditions of the required optimization process or application of the Best Available Technique, or BAT, principle.

Practice-related factors

Whether the waste is a result of an on-going conscious activity or not
Exposure from historic waste, created at a time with rules different from those of the present, and waste from an accident are both examples of what ICRP has termed existing exposure.

Whether the waste comes from a regulated process of not
Processes in society with radioactive material are subject to the Radiation protection law, but there is a difference between activities involving NORM, and NORM wastes, and practices involving radioactive material in a hospital or nuclear power plant. The latter has work arranged in a regulated way and radiation protection measures can be taken. To go from an unregulated to a regulated process involves several steps, including issues such as making retrospective judgement of justification which might lead to a change in the practice producing the waste.

Whether the waste production is incompletely known or not
Sometimes the owner of the waste may not even be conscious of its radioactive properties. This is the case in Sweden for water filters in water systems connected to private wells. An activity such as producing drinking water cannot easily be stopped and regulation cannot be made as for an activity that can be changed or made to stop. In certain cases a change in the NORM production can be induced given time information and resources, so that fewer measures need to be taken. These factors also need attention.

Whether a stream is temporary or continuous
Limitation in the final disposal must consider whether a waste stream represent a limited waste component whether the task is to regulate a continuous stream for years to come.

Whether a waste stream can be lead into alternative waste disposal facilities
If there for a certain waste stream is a choice between different repositories with different protective capability, the choice of repository can represent an optimization for different components in the waste stream. Also optimization may be possible in connection with placing waste within a single repository, in different compartments, or by placing waste high or low.

Properties of the waste of the repository

The properties of the wastes' radioactive substances, e.g. radiotoxicity and half-life
The properties of the radioactive substances are important in exposure scenarios. The waste's radiotoxicity, half-life and chemical form must be taken into account. The waste may also appear as a contamination of another type of waste, the chemical form of which must also be taken into account.

The total amount and concentration of radioactive substances in the waste
Activity concentration is an obvious factor that must be taken into account. A small strong source in the middle of a disposal site cannot be equated with a large number of low activity sources, even if the total activity is the same.

Potential intrusion
Human intrusion or human action – usually restricted to unintentionally intrusion – is often used as a well-defined term. ICRP Publication 81 mentions intrusion and gives criteria in terms of dose rate after intrusion – 10 to 100 mSv/a - for the need to consider improvement in repository design to mitigate intrusion consequences. Intuitively, the probability of intrusion is related to the strength of the barrier system and measures taken for institutional control. We should observe firstly, that the range of dose rates given by ICRP allows for a graded approach for different wastes and barrier systems and secondly, that intrusion into waste disposed of in a municipal disposal site after clearance should not be included in the intrusion category at all, but rather be considered to be part of the natural development, with some caveats. A separate criterion is given in section 4 in connection with intrusion into such a disposal site.

Problems in the attempts to avoid intrusion must not obscure our efforts of conservation of important societal environment-related information. Such a system cannot be said to guarantee protection in society in the future, but work in this area should be central within our obligation to keep doses to individuals in the future As Low As Reasonably Achievable, ALARA, in accordance with the optimization principle. In this connection, it is worth while to mention the Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management under the auspices of IAEA. As a result of the recurring reporting activities foreseen by this Convention, a build-up of information at the Agency can be expected with time so that a de facto – if not a formal - archive of waste management, and waste disposal information will be established, as a valuable addition to national archives that may come to exist.

4 Boundary conditions in terms of dose

Exempted or cleared materials, need by definition not to be regulated. The same is true for Release to air or water, and once it is established that releases are below regulatory requirements, society does not require detailed information after the discharge, contrary to the case of deposited material in a licensed facility for radioactive waste disposal. If waste is disposed of in a deposal sited as a result of conditional clearance, society has knowledge of the waste. The fact that there is no radiation protection regulation imposed after the stipulated conditions have been met, does not imply that society has abandoned all its rules. It merely implies that the waste is controlled by more general rules, such as - in the case of conditional clearance in a municipal disposal site - the institutional control on

the site, control of drinking water quality, and municipal records for future municipal planning purposes.

Information conservation, mentioned earlier, is an important necessary protection measure, but it does not guarantee protection for all future. For municipal or other near surface disposal sites, regulation should ensure that loss of information does not imply a drastic change of the radiation environment.

Assuming consideration has been given to the harmonization factors above, some reference for individual annual dose (seen as an order of magnitude) is suggested:

- *10 microsievert* . This dose implies that there is normally no further need for supervision from the point of view of radioprotection.
- *100 microsievert.* This represents a burden on future societies which constitutes a reason for regulation. It does however not necessarily constitute an undue burden.
- *1 mSv.* This represents a dose that can be accepted in an exceptional situation of intrusion in a municipal disposal site. That would guarantee that no individual from the public would suffer a drastic change in risk if society's supervision should fail. The intrusion dose range 10 – 100 mSv/a mentioned in ICRP81, should be reserved for repositories protected by stronger barrier systems and higher degree of institutional control.

5 Nordic common view

There is a large amount of international discussion in the field of non-nuclear and NORM waste management. It is the author's opinion that this discussion must have in focus the basic principles of disposal such as those discussed in this paper and that a Nordic common view would be welcome.

Clearance of decommissioning waste by measurements on samples

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Abstract

The nuclear research reactors and the Hot Cell facility at the Risø site have all been closed and are in the process of being decommissioned. The generated decommissioning waste contains objects (or parts) that are candidates for clearance and for which activity concentration measurements will be made. The Clearance Function in Danish Decommissioning is responsible for the measurements and for the demonstration of compliance with the clearance levels. Clearance measurements include measurements of surface-specific or mass-specific activities and can be made on objects as a whole or on samples from an object. The latter method is the focus of this paper. Clearance measurements on samples are performed in two ways (A and B) depending on the activity distribution in the objects. Procedure A is used for objects with inhomogeneous distribution of activity in which at least the location of maximum activity concentration is known. Procedure B is used for objects with an almost homogeneously distributed activity. The method has been validated by the Danish accreditation body DANAK in conjunction with the granting of an accreditation to the Clearance Function according to the international standard ISO/IEC 17025:2005. The paper presents how compliance with clearance levels is demonstrated for procedures A and B. This involves situations where several radionuclides are present in a sample, where an activity concentration is at or below the critical value (in a measurement) and where an activity concentration is calculated by scaling from the activity concentration of another radionuclide.

Introduction

The nuclear facilities at the Risø site are in the process of being decommissioned. The state company Danish Decommissioning is responsible for this task [1]. At present the small DR 1 reactor (2 kW) is fully decommissioned, the DR 2 reactor (5MW) is almost decommissioned and some exterior systems at the DR 3 reactor (10 MW) have been decommissioned. Next in line are the Hot-Cell facility followed by the interior parts of the DR3 reactor and eventually the Waste Treatment Plant. The goal for the decommissioning projects is green field status for the remaining buildings and land.

An important part of the decommissioning process is the separation of waste into radioactive waste and waste that can enter the conventional waste stream. This has prompted the establishment of an infrastructure for clearance measurements: *The Clearance Function*. The Clearance Function has developed procedures for clearance measurements [2] and it performs most of the measurements. The Clearance function includes laboratory facilities, measuring equipment, waste handling software, software for clearance related calculations and trained personnel.

In June 2007 the Clearance Function obtained an accreditation from the Danish accreditation body, DANAK [3] according to the international standard ISO/IEC 17025:2005. DANAK is a member of ILAC, the **I**nternational **L**aboratory **A**ccreditation **C**ooperation. The operational framework for the Clearance Functions is described in a quality manual. The manual includes two procedures for clearance measurements on objects by the use of measurements on samples from the objects. The paper will focus on these procedures.

Clearance and clearance criterion

Objects/materials can be released from regulatory control when the activity concentration of a radionuclide i in the object, C_i, is below the clearance level, CL_i, for that nuclide *i.e.*:

$$\frac{C_i}{CL_i} \leq 1 \tag{1}$$

If more than one radionuclide is present in the object the general clearance criterion apply:

$$\sum_{i=1}^{N} \frac{C_i}{CL_i} \leq 1 \tag{2}$$

The activity concentrations C_i are mass-specific activities in the case of bulk contamination of the object, or surface-specific activities in the case of surface contamination; mass-specific and surface-specific clearance levels must be used accordingly. The activity concentrations, C_i, are average values. The Danish radiation protection authority has adopted the recommended clearance level values from IAEA (mass-specific clearance) [4] and from EU (surface-specific clearance) [5]. Averaging of activity concentrations can be made only for masses up to 1,000 kg and areas up to 1 m^2.

As activity concentrations are assessed by measurements, uncertainties are introduced and absolute compliance with Eq. 2 can not be made. Consequently, the Danish radiation protection authority has specified that it must be ensured that released objects have at least a 95% probability of fulfilling Eq. 2. For measurements on total objects (*in toto* measurements) this can be achieved by requiring that measured concentrations, Cm, and the associated standard uncertainties, $u(Cm)$, fulfil the following criterion:

$$\sum_{i=1}^{N} \frac{Cm_i}{CL_i} + 1.65 \cdot \sqrt{\sum_{i=1}^{N} \frac{u_i(Cm)^2}{CL_i^2}} \leq 1 \tag{3}$$

The summations in Eq. 3 must be made for all radionuclides present or that can be present in the object. For radionuclides that can be present but show no measurable activity, the critical value of activity concentration [6] determined by the measurement conditions is used as activity concentration and the uncertainty of determining the critical value is used as uncertainty. For bulk contamination, *in toto* measurement are made by gamma-spectroscopy. If an object contains non-gamma-emitting radionuclides, the contents of these radionuclides must be accounted for in a different way. This can sometimes be done by scaling to one of the gamma emitters in the object, if the activity ratios are known (nuclide vector). In these cases scaled activities are included in Eq. 3.

Clearance using measurements on samples

In some cases the most convenient way (or the only feasible way) to get information on the average activity concentration in an object is to measure the activity concentration in samples from the object. Two procedures, A and B, have been developed for such cases. The procedures are described for mass-specific activities but with proper transformation of terms they can be used for surface-specific activities as well.

Procedure A (relative activity concentration is known)

This procedure can be used when the relative variation of the activity concentration in an object is known to such an extent that the object can be divided into regions and samples can be taken from

parts of these regions where the concentrations are highest. Such information was obtained for the concrete biological shield around the DR 1 and DR 2 reactors.

The measured concentration from a (conservative) sample from a region is regarded as the concentration in that region. From the measured activity concentrations, $Cm_{i,j}$, of nuclide i, in a sample from region j, which has the mass m_j, the clearance criterion becomes:

$$\sum_{i=1}^{N} \frac{\dfrac{\sum_j Cm_{i,j} \cdot m_j}{\sum_j m_j}}{CL_i} + 1.65 \cdot \sqrt{\sum_{i=1}^{N} \frac{u\left(\dfrac{\sum_j Cm_{i,j} \cdot m_j}{\sum_j m_j}\right)^2}{CL_i^2}} \qquad (4)$$

The u-terms in the criterion are the propagated uncertainties of the average activity concentrations in the object. A region can be assigned a value measured in another region if that region is known to have a higher activity concentration. If the clearance criterion is fulfilled for the regions with the highest activity concentration, the object can be released.

The maximum mass for averaging activity concentration is 1,000 kg, hence objects of larger masses must be treated in a way that ensure that any arbitrary 1,000 kg division of the object fulfil the clearance criterion for the object to be released. The clearance criterion ensures that released objects have a minimum 95 % probability of fulfilling Eq. 2.

Example using procedure A

A block of concrete from the biological shield of a reactor was a candidate for clearance. The activity in the block was caused by neutron activation and from the knowledge of the neutron field around the reactor core the pattern of activation in the block is known. Only ^{60}Co and ^{3}H were relevant activation products. The ^{3}H concentrations were derived from the ^{60}Co concentrations by multiplying by 1,000.

The most conservative cut (maximising activity concentration in 1,000 kg) was envisaged, and from this piece, four regions were sampled. The results of the measurements are shown in Table 1.

Table 1. *Measurement results for 4 samples. Standard uncertainties are only given for activity concentrations in the entire piece.*

Sample/region	Mass of region [kg]	^{60}Co concentration in sample [Bq/g]	^{60}Co concentration in entire piece [Bq/g]	^{3}H concentration in sample [Bq/g]	^{3}H concentration in the entire piece [Bq/g]
1	400	0.012	0.026 ± 0.008	12	26 ± 12
2	200	0.022		22	
3	150	0.031		31	
4	100	0.080		80	

Inserting the numbers of the activity concentrations in the left side of Eq. 4 gives:

$$\frac{0.26}{0.1}+\frac{26}{100}+1.65\cdot\sqrt{\frac{0.008^2}{0.1^2}+\frac{12^2}{100^2}}=0.76$$

Since this value is less than 1, the entire block of concrete can be released.

Procedure B (almost homogeneous activity concentration)

This procedure can be used when the activity concentration in an object is fairly homogeneous throughout the object without any hot-spots. This has been the case for some cooling circuits at the DR 3 reactor. In these circuits water has circulated for a long time and each part of the inner surfaces have experienced the same flow of water (have had "the same history"). As the activity concentration is almost homogeneous no maximum mass for averaging exists in this procedure.

The clearance criterion given by Eq. 2 is tested by the use of activity concentrations in a number of samples taken from the object. The test is done by testing the hypothesis:

$$\sum_{i=1}^{N}\frac{C_i}{CL_i}\geq 1 \tag{5}$$

If this hypothesis is tested false, the object can be released. The *Sign-test* [7, 8] is used to make the test. The Sign-test can be used in cases where background levels of radioactivity in the samples are not to be subtracted and the test is independent of the (small) variation in activity concentrations in the object.

A type I error is made if the hypothesis is rejected when Eq. 5 is true (*i.e*. an object is released when it should have been withheld), and a type II error is made if the hypothesis is not rejected when Eq. 5 is false (*i.e*. an object is withheld while it could have been released).

The minimum number of samples, N_p, necessary to decide whether the hypothesis must be rejected or not depends on the following:

1. The estimated value of the test parameter Y, calculated from estimated values of activity concentrations, $Cest_i$:

$$Y=\sum_{i=1}^{N}\frac{Cest_i}{CL_i} \tag{6}$$

2. The estimated propagated standard uncertainty on the parameter Y, $u(Y)$.
3. The maximum probability for making type I errors and the desired maximum probability for making type II errors, when this is regarded practical and economical feasible. This determines the width, Δ, of the so-called grey region [6]. In procedure B the probability for making type I errors is set to 5 % and the probability for making type II errors is maximised to a maximum error probability of 5 %. This maximization makes $\Delta = 1-Y$.

Given the above probabilities, the minimum number of samples, N_p, can be calculated as (rounded upwards to the nearest integer):

$$N_p = \frac{10.824}{4 \cdot \left(\frac{1}{\sqrt{2\pi}} \int_{-\infty}^{\frac{\Delta}{u(Y)}} \exp\left(\frac{-x^2}{2} \right) \mathrm{d}x - 0.5 \right)^2} \tag{7}$$

Samples can now be taken. It is important that the positions of the samples are distributed fairly even in the object. When the samples have been measured, the Y_j-values for each sample j must be calculated from the measured activity concentrations in each sample, $Cm_{i,j}$:

$$Y_j = \sum_{i=1}^{N} \frac{Cm_{i,j}}{CL_i} \tag{8}$$

If no activity is measured for a nuclide that might be present in the sample, the critical value of the activity concentration is used instead.

The conditions for calculating the minimum number of samples must be checked. This is done by checking that the mean value of the Y_j-values is lower or equal to the estimated test parameter Y and that the experimental standard deviation of the Y_j-values is less or equal to the estimated standard uncertainty on the Y-parameter. Is this not the case a new minimum number of samples must be calculated using the new estimates of Y and $u(Y)$ and additional samples must be obtained and measured and the conditions validated. Again, the condition with no hot-spots must not be violated.

Having taken and measured N_p, the number of Y_j's, which are below 1 is counted and called $S+$. The number of times (if any) that Y_j's equals 1 is also counted and subtracted from N_p to give $N_{p,adj}$. If $S+$ equals $N_{p,adj}$ the hypothesis of Eq. 5 is rejected and the object can be released. If $S+$ equals 0 the hypothesis can not be rejected and the object must be withheld. If $1 < S+ < N_{p,adj}$, the $S+$ value must be compared to a critical figure k_c [7] for the Sign-test, which is a function of $N_{p,adj}$ and the selected maximum error percent for type I errors (in this case 5 %). Only if $S+ > k_c$ the hypothesis of Eq. 5 can be rejected.

Example using procedure B

Consider a part of a secondary reactor cooling system. In the system water (H_2O) had circulated removing heat from the primary cooling system (D_2O) through a heat exchanger. The heat exchanger had a leak and small amounts of primary cooling water passed into the secondary cooling system. The secondary system had its water content replaced at a regular basis. The water had been drained and the piping dismantled. An inspection of the inner surfaces of the pipes revealed an almost uniform layer of corrosion of a few millimetres, hence the piping was considered to have "the same history". The outer surfaces of the pipes were not contaminated.

From measurements on primary and secondary cooling water it was expected that only ^{3}H (clearance level = 100 Bq/g) and ^{14}C (clearance level = 1 Bq/g) could be present in the pipes. From preliminary measurements the average mass-specific activity of ^{3}H was estimated to be 40 Bq/g with a standard uncertainty of ≈ 10 Bq/g. The average mass-specific activity of ^{14}C was estimated to be ≈ 0.3 Bq/g with a standard uncertainty of ≈ 0.15 Bq/g. These estimates resulted in the following parameter values, Y = 0.7, Δ = 0.3 and $u(Y)$ = 0.2. With these parameter values, Eq. 7 gives 15 samples (as the minimum).

Fifteen samples were taken from the pipes sampled at regular intervals along the piping. The samples were obtained by drilling through the pipe wall from the outside. The samples were analysed by

gamma spectroscopy to assist the assumption that only ^{3}H and ^{14}C were present in the pipes. The gamma measurements were so lengthy that the critical values of activity concentrations of relevant radionuclides were at a very small fraction of the clearance levels and could be ignored. Each sample was dissolved and analysed by scintillation counting. The analyses revealed only a content of ^{3}H and ^{14}C. The measured concentrations and *Y*-values are listed in Table 2.

Table 2. *Measurement results for fifteen samples from the pipes of the secondary cooling system.*

Sample	^{3}H concentration [Bq/g]	^{14}C concentration [Bq/g]	Y_j
1	35	0.21	0.56
2	37	0.23	0.60
3	35	0.27	0.62
4	41	0.24	0.65
5	45	0.65	1.10
6	55	0.49	0.80
7	27	0.26	0.60
8	35	0.40	0.85
9	41	0.27	0.68
10	32	0.22	0.54
11	42	0.25	0.67
12	37	0.10	0.47
13	30	0.15	0.45
14	55	0.45	1.00
15	27	0.20	0.47
Mean concentration ± experimental standard uncertainty	**38 ± 9**	**0.27 ± 0.13**	**0.65 ± 0.19**

The measurements show that the observed mean concentrations and uncertainties are below the estimated values and thus no further samples are needed. The *S*+ value is 13. The critical value for the adjusted minimum sample number ($N_{p,adj}$ = 14) is 10. As the *S*+ is greater than the critical value the hypothesis Eq. (5) is rejected and the pipes of the secondary cooling system can be released.

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Clearance of buildings and land after decommissioning of nuclear facilities at the Risø site

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Abstract

The nuclear installations at Risø National Laboratory have all been closed for decommissioning. The DR 1 reactor was closed in 2001 and the decommissioning began in 2004. It was planned to use the reactor building for other purposes and therefore clearance measurements for both the reactor building and the surrounding land were needed to document an unconditional clearance of the reactor site.

The clearance condition for buildings were the nuclide- and surface-specific clearance levels recommended by the European Commission and derived from a dose criterion of 0.01 mSv/a, and exposure scenarios in which the buildings are used for other purposes. Clearance levels should be applied to the total activity on and below each single square meter of the building area.

International recommended clearance levels for land areas do not exist. Therefore, such levels had to be derived. This was done using similar considerations as for the buildings, *i.e*. a dose criterion of 0.01 mSv/a, and exposure scenarios for future use of the area. As the future use of the land area was unknown, clearance levels were derived from combined external and internal exposure pathways assuming that milk, meat, vegetables and grain products were produced and consumed within the area. Clearance levels for ^{60}Co, ^{90}Sr and ^{137}Cs were determined to be 3,000 Bq·m^{-2}, 11,000 Bq·m^{-2}, and 7,000 Bq·m^{-2}, respectively.

The measurement strategy and the measurement results for the DR 1 reactor building and the surrounding land area are presented. Both gross-β and γ-spectrometric measurements were used. It is shown how the measured values and the associated uncertainties were used to demonstrate compliance with the clearance levels.

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 all its storage facilities for radioactive waste. All the nuclear facilities except the Waste Management Plant have been closed. 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.

Different types of measurement are used for buildings and land. A combination of gross α-/β-measurements, *in situ* γ-spectrometric measurements, and laboratory analyses are used for land allowing depth distributions of γ-emitters to be determined. Direct measurements of α-/β-contamination are often adequate for building surfaces unless the radionuclides have penetrated into the materials.

The rooms in the building and the surrounding land were divided into classes according to their potential of being contaminated. Measurements inside the building and on the surrounding land were performed mainly with high-sensitive ZnS contamination monitors but also with high-efficiency Ge-detectors.

Surface-specific clearance

The process of clearing buildings and land involves the measurement of extensive surface areas. Since the buildings at the Risø-site are only slightly contaminated with radionuclides it should, in most cases, be possible to clear them after they have been cleaned or decontaminated. The criteria for clearing buildings or the resulting building rubble and land must ensure that the 0.01 mSv/a dose criterion can be respected. In applying clearance criteria it might be advantageous to divide the buildings and land into classes depending on the extent of expected contamination. For example 100 % of a floor of a controlled area might be measured while the ceiling probably would involve a procedure where only, say, 10 % of the surface needs to be measured.

Clearance criteria for buildings

A cleared building can be left standing and used for other non-nuclear purposes or it can be torn down. EU has developed exposure scenarios for evaluating the radiological impact of clearing buildings for continued use [1, 2]. The strategy used was to identify the most restrictive scenario for each of the exposure pathways: external irradiation, inhalation, ingestion and skin contamination, and then use the scenarios to derive the clearance levels by setting the level such that the resulting dose equals 0.01 mSv/a. The radiological analysis for building reuse also takes into account the scenarios which are developed for clearance of buildings for demolition. The recommended clearance levels from EU and the most restrictive exposure scenario are shown in Table 1 for a few selected radionuclides.

Table 1. *Clearance Levels (CL) for building reuse as recommended by the European Commission and the exposure scenario from which these levels have been derived.*

Radionuclide	Most restrictive Exposure scenario	Clearance Level [Bq·m^{-2}]
^{60}Co	External radiation	10^4
^{90}Sr	Intake of vegetables	10^6
^{137}Cs	External radiation	10^4
^{152}Eu	External radiation	10^4
^{238}U	Inhalation	10^4

In almost all practical situations more than one radionuclide is present in the surface contamination. In such cases the clearance criterion is met when:

$$\sum_{i=1}^{N} \frac{C_i}{CL_i} \leq 1$$

where C_i is the surface contamination for nuclide i and N is the number of radionuclides present.

Clearance criteria for land

Criteria for clearance of land have not been recommended internationally. Therefore, nuclide specific criteria have been developed from a dose criterion of 0.01 mSv/a to the most exposed persons (representative persons). A more restrictive approach has been used, compared to that for deriving clearance criteria for buildings [1, 2]. It is assumed that the exposure scenario include both external

irradiation from the land as well as internal irradiation from ingestion of foodstuffs grown at the land [3].

The external exposure is calculated for a surface area of 1,000 m^2 where the activity is homogeneously distributed in the upper 25 cm layer and for an indoor/outdoor occupancy of 80 %/20 % in a house situated in the centre of the surface. A time-averaged shielding factor of 0.2 has been used in the exposure calculations to account for this occupancy pattern and an exposure time of 8,760 hours per annum has been used. The surface contamination causing an annual external dose of 0.01 mSv is shown in Table 2 for a few radionuclides.

The internal exposure is calculated for consumption of milk, beef, vegetables and grain products, which are assumed to be produced within the area. Furthermore, it is assumed that 10 % of a normal Danish consumption of these products originates from the area. An average annual food consumption of the products has been used and the transfer factors applied have been determined for Danish soil conditions. The calculated surface contamination density causing an annual internal dose of 0.01 mSv from the food consumption (10 % of total) is shown in Table 2 for a few selected radionuclides.

Table 2. *Clearance Levels for land derived from a combined external exposure from γ-radiation and internal exposure from ingestion of locally grown foodstuffs.*

Radionuclide	Surface contamination per external dose, C_{ex} [Bq·m^{-2}/0.01 mSv·a^{-1}]	Surface contamination per internal dose, C_{in} [Bq·m^{-2}/0.01 mSv·a^{-1}]	Clearance Level, *CL* [Bq·m^{-2}]
^{60}Co	$3.0 \cdot 10^3$	$5.7 \cdot 10^5$	$3 \cdot 10^3$
^{90}Sr	-	$1.1 \cdot 10^4$	$1 \cdot 10^4$
^{137}Cs	$1.1 \cdot 10^4$	$1.6 \cdot 10^4$	$7 \cdot 10^3$

The nuclide-specific clearance level for nuclide *i* for the combined external γ-exposure and internal β-/γ-exposure has been determined as:

$$CL_i = \frac{1}{\frac{1}{C_{ex,i}} + \frac{1}{C_{in,i}}}$$

The calculated clearance levels, *CL*, are shown in Table 2 for a few selected radionuclides. It should be emphasized that these levels are rather conservative as they are derived for combined exposure pathways and 100 % occupancy at the site (80 % indoor and 20 % outdoor). An occupancy pattern where working hours are spent away from home would probably triple the clearance levels for those radionuclides for which the external γ-dose is the decisive component, *e.g.* ^{60}Co.

Compliance with clearance criteria

The condition for clearance of a building or land surface is that the surface-specific clearance levels should be applied to the total activity on and below the surface, divided by the surface area, *i.e.* to the sum of fixed and non-fixed activity on the surface plus the activity that has penetrated into the material. Averaging of the surface contamination density (surface contamination) is allowed only square meter by square meter of the total surface area. In addition, the sum over all measured nuclides,

i, of the ratios of the nuclide-specific surface contamination, C_i, on the surface to the corresponding clearance level, CL_i, should be less than or equal to 1:

$$\sum_{i=1}^{N} \frac{C_i}{CL_i} \leq 1$$

Due to uncertainties in the measurement of surface contamination it is necessary to include these uncertainties. Assuming that the measured nuclide-specific surface-concentrations, C_i, have the standard uncertainty, $u(C_i)$, the clearance index, CI, is defined as:

$$CI = \sum_{i=1}^{N} \frac{C_i}{CL_i} + 1.65 \cdot \sqrt{\sum_{i=1}^{N} \frac{u(C_i)^2}{CL_i^2}}$$

where N is the number of radionuclides. The second term of the CI is the combined standard uncertainty of the first term multiplied by a coverage factor of 1.65. This gives a high probability (at least 95%) that the quantity $\Sigma_i C_i/CL_i$ is less than 1, if the clearance index, CI, is less than 1.

If the contamination is situated only on the surface, measurements with an α-/β- contamination monitor will be sufficient. The clearance index, CI, can for each single square metre be expressed as:

$$CI = \frac{\overline{C}_\alpha}{CL_\alpha} + \frac{\overline{C}_\beta}{CL_\beta} + 1.65 \cdot \sqrt{\left(\frac{u(\overline{C}_\alpha)}{CL_\alpha}\right)^2 + \left(\frac{u(\overline{C}_\beta)}{CL_\beta}\right)^2}$$

where $\overline{C}_\alpha$ and $\overline{C}_\beta$ is the average α- and β-surface contamination over one square metre and $u(\overline{C}_\alpha)$ and $u(\overline{C}_\beta)$ are the corresponding standard uncertainties. The average surface contamination density (α- and β-contamination) is given as:

$$\overline{C} = \frac{1}{N_{tot}} \cdot \sum_{i=1}^{N_{\text{tot}}} \left(C_i - C_{back,i}\right)$$

where $C_{back,i}$ is the measured background contamination and N_{tot} is the number of surface contamination measurements over one square metre. The standard uncertainty of the measured average surface contamination, $u(\overline{C})$, is given as:

$$u(\overline{C}) = \frac{1}{N_{tot}} \cdot \sqrt{\sum_{i=1}^{N_{tot}} \left(u(C_i)^2 + u(C_{back,i})^2\right)}$$

where $u(C_i)$ and $u(C_{back,i})$ is the standard uncertainty of each surface contamination measurement and of each background measurement within one square metre, respectively.

Clearance measurements on DR 1 building and surrounding areas

From a regulatory viewpoint, it is necessary to be able to verify compliance with the clearance levels. This can be done by direct measurements, *e.g.* direct surface measurement, in situ gamma spectrometry, statistical sampling and by laboratory measurements on representative samples.

In the case of building structures, the surface-specific clearance levels apply to the total activity in the structure, *i.e.* the sum of removable and fixed activity. When applying this concept of total activity in the structure per unit surface area, the penetration depth must be taken into account when clearance

measurements are carried out, *e.g.* by properly choosing the calibration for in situ gamma spectrometry.

In the case of land, the surface-specific clearance levels also apply to the total activity at and below the surface. For undisturbed soil at and around the Risø-site measurements of fall-out ^{137}Cs from nuclear weapons testing have shown that the depth distribution in soil is exponential and this distribution was applied in the calibration of the germanium detectors.

Classification of buildings and land

The DR 1 building and the areas surrounding the building were classified according to their potential of being contaminated or activated:

- Class 1 areas: Areas with a small probability of being contaminated or activated by neutrons to a level higher than the clearance levels. The measurement coverage of these surfaces is 100%.
- Class 2 areas: Areas with a large probability of being contaminated or activated by neutrons to a level below the clearance levels. The measurement coverage of these surfaces is 10 - 50%.
- Non-classified areas: Areas where there has been no or very little contact with radioactive materials. Only a few random measurements of these surfaces will be made to verify the classification.

Most of the areas were classified as class 2 areas with a few class 1 areas, including the reactor hall.

The asphalt surface north of the building was classified as a class 1. Towards south, west, and east there are meadows and cultivated fields being regarded as non-classified areas.

Clearance measurements on buildings and land

In the reactor hall contamination of the walls, the ceiling and the floor was measured with germanium-detectors and α-/β-contamination monitors.

Figure 1. *Clearance measurements performed inside the DR 1 reactor hall. The left-hand picture shows γ-spectrometric measurements of the "crater" and the right-hand picture shows contamination measurements of the floor with an α-/β-contamination monitor.*

The area left open after the removal of the reactor block, the "crater", was measured with a germanium-detector. This surface was activated, and the activity was depth-distributed into the concrete. Drill samples had revealed that the activity concentration was decreasing exponentially as a function of depth and that the relaxation length was about 10 cm. This value was used to calculate the total activity below a surface of 1 m^2.

Figure 2. *Clearance measurements on the area around the reactor building using a germanium-detector for measurements on asphalt and grass surfaces.*

It was important to distinguish the ^{137}Cs originating from fall-out from nuclear weapons testing, fall-out from the Chernobyl accident and potential ^{137}Cs-contamination originating from the reactor operation. A background spectrum was therefore measured in a private home, where the detectors viewed a landscape similar to the one surrounding reactor DR 1. The background measurements for the ceiling were performed in a workshop at the Risø-site. The construction design of this workshop is similar to the architecture of the DR 1 building.

The clearance measurement programme showed that no contamination above the clearance levels was found in the DR 1 building and no contamination originating from the operation of the reactor was found on the surrounding land [4].

Conclusions

Clearance measurements on building and land require sensitive equipment to be able to detect surface contamination at and below the clearance levels with the necessary accuracy. Furthermore, clearance measurements are rather time consuming and the personnel carrying out the measurements needs specific training in order to verify that the measuring data can comply with the clearance levels. The programme of clearance measurements on reactor buildings and surrounding land was performed over a period of about three months and required a total manpower of around six man·months. The programme and its results were documented in a final survey report [4].

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New Norwegian repository for LSA scale from the petroleum industry

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Abstract

The Norwegian petroleum industry faces problems with low specific activity (LSA) scale (also called TE-NORM) settling inside pipes and other production equipment. About 400 tons of scale are stored at different dedicated locations. It is expected that the annual production of scale will increase from the present 20-30 tons to more than 100 tons in the future. The production increases with increasing age of the oil fields.

An Environmental Impact Study for a planned repository site in Gulen municipality at the western coast of Norway was presented to the authorities in November 2004 and was approved. The acceptance from the local community and the public, as well as from the oil industry, was favorable.

The construction of the repository in Gulen started in August 2006, and was completed in the autumn of 2007. It has a capacity of 6300 tons of LSA scale and may be expanded if necessary. The scale will be cleaned for oil, dried and packed in HDPE-lined concrete containers before disposal. The Government has given a State Fund Guarantee for the repository. The permit of operation was granted in March 2008. The repository is ready for operation in the last half of 2008.

LSA scale

LSA scale is a radioactive deposit inside pipes and other production equipment and consists of carbonates and sulphates of Ca, Sr, Ba and co-precipitated Ra. The salts were dissolved in the reservoir itself in a mixture of original formation water and injected seawater. When transported to the surface together with the oil, the pressure and temperature drops, and the salts are deposited. LSA scale is a type of NORM, often specified as TE-NORM – technically enhanced naturally occurring radioactive material.

The annual production in Norway of LSA scale is about 20 tons. It is expected to increase to more than 100 tons within few years, as the production increases with increasing age of the oil fields. The stored quantities of scale, sludge and similar materials are about 400 tons at different coastal bases, containing a total of about 0.25 gram of radium. As radioactive waste LSA scale may be classified as very low level waste (VLLW).

In connection with the construction of the national combined disposal and storage facility for low and intermediate level radioactive waste (KLDRA) in Himdalen, the Norwegian Radiation Protection Authority (NRPA) decided that the scale, because of its large volume, should not be disposed of in the KLDRA. NRPA therefore defined it as *non-nuclear* radioactive waste. The petroleum industry should therefore have to find other solutions for disposal of the scale.

Final solutions for LSA scale

In common terms the word "repository" means "facility for final (eternal) storage (of waste)". The radioactive half-life of ^{226}Ra, which is the most long-lived radioactive component of LSA scale, is 1600 years. Thus for a scale repository "eternity" is about ten half-lives of ^{226}Ra.

A repository for LSA scale should satisfy the following criteria:

- The waste material should be chemically stable.
- The repository should withstand weathering.
- The repository including the surrounding area should have a sustainable owner.
- The repository concept should be likely to be accepted by future generations.

There are three examples of final storage of LSA scale in Norway and UK:

- Storage of 70 tons of LSA scale in the KLDRA Himdalen (Norway), which is a repository for low- and intermediate level radioactive wastes.
- Storage of material from 44 drums from Brent Spar at Drigg (UK).
- Storage of unknown amounts of LSA scale in a "non-constricted repository" in UK: the bottom sediments of Aberdeen harbour.

As waste, LSA scale is handled according to the same principles as all other waste in the industry i.e. in a "cradle to grave" perspective with focus on highest possible degree of reuse and recycling. Therefore, LSA scale contaminated equipment is cleaned by e.g. high-pressure water jetting, facilitating recycling of the component steel and minimisation of the NORM waste.

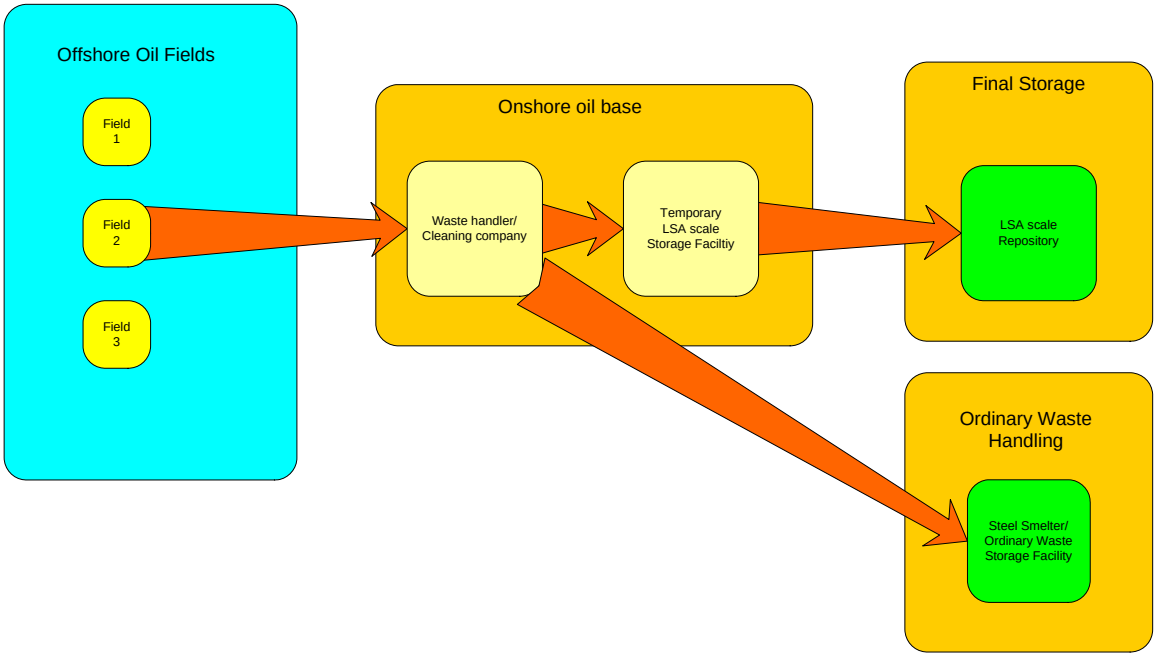

Figure 1. *Cradle to grave perspective of LSA scale.*

The new repository

In connection with the construction of the national combined disposal and storage facility for low and intermediate level radioactive waste (KLDRA) in Himdalen, the Norwegian Radiation Protection Authority (NRPA) decided that LSA scale, because of its large volume, should not be disposed of in the KLDRA. The petroleum industry should therefore have to find other solutions for disposal of the scale.

An *Environmental Impact Study* for a planned repository site in Gulen municipality at the western coast of Norway was presented to the authorities in November 2004 and was approved. The acceptance from the local community and the public, as well as from the oil industry, was favorable.

The Gulen municipality is located about 70 km north of the city of Bergen. The repository is constructed in a quarry area on the northern side of the Fen Fjord. The Mongstad refinery, owned by Statoil, is located on the southern side of the fjord.

Figure 2. *Location of the new repository, called Stangeneset at the map. The black arrow at the photo points at the repository entrance. The Mongstad refinery is seen in the far background of the photo, across the fjord.*

The construction of the repository in Gulen started in August 2006, and was completed in the autumn of 2007. It has a capacity of 6300 tons of LSA scale, which may be expanded if necessary. The Government has given a State Fund Guarantee for the repository, similar to the one given for the KLDRA facility, and has thereby approved it. Due to an explosion in May 2007 in a large oil tank in the close vicinity of the repository, the authorities did not want to grant a permit of operation until official accident investigations had been performed and conclusions drawn. However, the Permit of Operation of the repository was granted by the NRPA in March 2008. The operation will start in the last half of 2008.

The operator of the repository is the company Wergeland-Halsvik AS. The company Norse Decom AS, which is a subsidiary of Institute for Energy Technology (IFE), is responsible for quality assurance, environmental monitoring and radiation protection.

The new repository is designed for safe disposal of NORM bearing waste from the oil and gas industry. The facility will receive untreated LSA scale which, after being conditioned and packed in HDPE-drums in concrete containers, will be disposed of in underground rock caverns. There the containers will be surrounded by injected concrete or clay on order to fill the empty space between the containers and the surrounding bedrock. Thus a waterproof zone is made between the waste and the environment.

Environmental monitoring programme

Radiological investigations were performed in 2005 and 2006, well before start of operation of the repository. Terrestrial samples of local rock, fly ash (from the fly ash landfill), birch, and bog cotton were collected at the surface area above the repository. Marine samples of sea water, sediment and

blue mussels were collected near the outlet of the discharge pipeline, while cod was sampled as close to the pipeline outlet as possible. Ground water was sampled from a drainage basin in the repository tunnel.

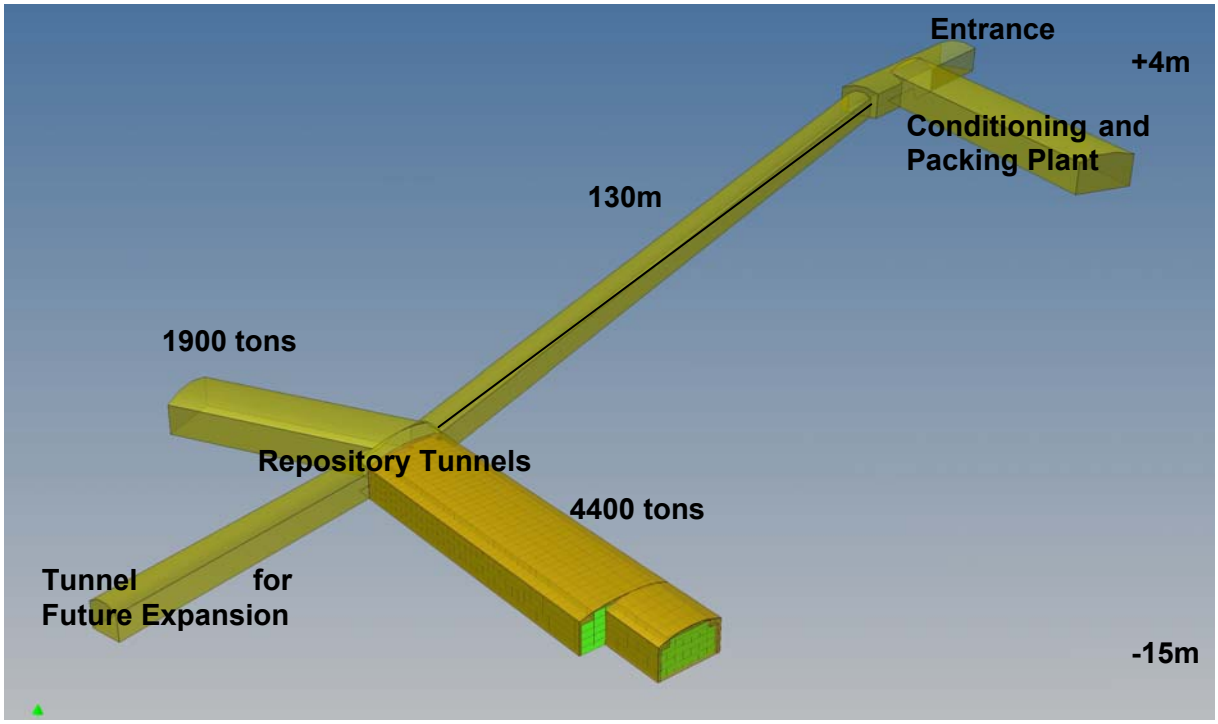

Figure 3. *A 3D-model of the repository, which is 235 m long, including the tunnel for future expansion. The 130 m long entrance tunnel leads from the conditioning and packing plant down to the repository tunnels 15 m below sea level.*

These radiological investigations will be repeated every year the first five years and every second year from then on to demonstrate that the repository does not contaminate the environment. In addition, samples of drainage and process water are analysed as a part of the annual environmental monitoring programme.

Contractual and legal conditions

The LSA scale producer is the scale owner. In practice this means the actual licence holding oil and gas company. Prices and other conditions for disposal in the new facility are defined in a major contract with Statoil (now StatoilHydro). However, all Norwegian producers have the right to establish contracts at the same terms. The LSA scale is defined according to current Norwegian legislation. Other conditions with respect to physical and chemical properties are also set. The country of origin defines the LSA scale "nationality". LSA scale can be exported from e.g. UK for disposal in the repository. The Gulen repository has the capacity. However, to do so requires approval from relevant authorities in *both* the exporting and the importing country.

Figure 4. *Photo of the largest repository tunnel.*

10 Nuclear Safety and Security and other on Radiation Protection

Presentation of the Swedish Radiation Safety Authority

Olaf Hallström

Swedish Radiation Protection Authority

A presentation of the background, the conditions for the merger of the Swedish Radiation Protection Authority and the Swedish Nuclear Power Inspectorate to become the Swedish Radiation Safety Authority and its newly formed organisation.

Radiation practice licence in Estonia

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Abstract

In Estonia first radiation practice licenses were given out in 1997 after Parliament accepted first Radiation Act. Radiation practice license is needed in order to have a right to do radiation practice, which is defined according to Radiation Act as any activity, which increase or may increase the exposure of humans to radiation emanating from an artificial source or from a natural radiation source in cases where natural radionuclides are processed in view of their radioactive, fissile or fertile properties.

During these 10 years the concept of licence, as well as licensing system have changed remarkably. In first years radiation practice licence was just one page document. Today it is several pages long document, which also includes special requirements. From 1997 until 2004 all licencing procedures were fullfilled by Estonian Radiation Protection Centre. In 2004 a new Radiation Act came into force in order to fullfil the European Union requirements. This Act introduced changed licencing system, where Ministry of the Environment is responsible for issuing the licence. This paper will give the overview of the developments in Estonia and tries also to explain the reasons for changes. Several solutions used in the process will be discussed as well.

At present preparation of national development plan for radiation protection is in the final stage. This document covers 4 different areas, including the infrastructure. The future aim is to provide easily understandable system of radiation protection, includin licensing for all sides.

International co-operation, especially with neighbouring countries was important factor in the development of the licencing procedure. This factor will be adressed too.

Paper

The Republic of Estonia is a small Baltic state that regained independence in 1991. Previous to that time, the Soviet system placed the use of radioactive sources under the supervision of the Sanitary Board. During the 1991 reorganization process, it was decided to separate responsibilities connected to ionizing radiation from general sanitation responsibilities. The primary reasons for this step were connected to the several “heritage” sites from the Soviet time: a nuclear submarine training centre with 2 reactors, uranium milling installations, mining tailing ponds and a radon-type radioactive waste depository for institutional waste.

The Estonian Radiation Protection Centre (ERPC) was established by governmental mandate in 1996, with a mission to oversee: the monitoring of radiation, laboratory analyses, licensing, radioactive waste management and export/import of radioactive source controls, participation in the preparation of legislation regarding radiation protection, prompt information exchange, and the fulfillment of the varied requirements of different international conventions. Additional duties have included consulting, and acting as a liaison point for the IAEA in Estonia.

Due to the political changes at the federal level, the Soviet legal base was no longer applicable. In order to prepare the legal basis for radiation protection, a working group was established in mid 1990s. The aim of the working group was to prepare an initial draft of the Radiation Act and the accompanying regulations. During that stage cooperation was an essential factor: the radiation acts of nearby Nordic countries were studied in order to learn from their experiences. The first Radiation Act was adopted by Parliament in 1997, which introduced the concept of the radiation practice licence as a document that is required in order to have a right to do radiation practice. Practice is defined as any activity, which increases or may increase the exposure of humans to radiation emanating from an artificial source or from a natural radiation source in cases where natural radionuclides are processed in view of their radioactive, fissile or fertile properties.

The first radiation practice licences were given out in 1997. During the first few years, the ERPC had the right to issue radiation practice licence and also had responsibility for enforcement. The Act of 1997 stated only some basic, primary requirements. For example: together with the application for the licence the following data had to be provided: information about actual and potential radioactive sources, practices, site, assessment of radiation safety, radioactive waste production assessment, information about additional protection measures and background regarding worker education and training. The Radiation Act also stated very concisely that issuing of the licence was possible when there were qualified personnel available and the site was suitable for suggested practice. The licenses were time limited, to a maximum of 5 years. According to legislation, the radiation practice licence was a single page document, containing the following information: issuer – Estonian Radiation Protection Centre; owner of the radiation practice licence, name of the radiation practice, validity and duration of the radiation practice licence. No special conditions were put down in the licensing procedure. Due to the fact that ERPC was a new institution, no hypothetical working procedure was in place to create a harmonised licensing approach. This may have too often placed the Commission in a position of reaction, rather than preventative action.

In 2001 the Environmental Supervison Act was revised and the enforcement rights and responsibilities regarding radiation practice licensing were given to Environmental Inspectorate. This created an appropriate distance between the licensing department and the enforcement agency, but this transition created numerous difficulties for the next several years. Already over mandated, and understaffed, the Environmental Inspectorate was not always able to fulfil the assigned responsibilities. Actual enforcement procedures came into play in 2005, working in co-operation with the ERPC.

Estonia joined the European Union in 2004. Amendments to the Radiation Act were necessary in order to bring the state in line with the relevant EURATOM Directives. Estonia introduced new version a Radiation Act, which is the main law covering radiation safety, in 2004 for this purpose. From 1997 until 2004 all licencing procedures were fullfilled by Estonian Radiation Protection Centre. Article 4 of Radiation Act specifies that the performance of activities related to radiation protection shall be organized by the Ministry of Environment within the limits of its competence through the Environmental Inspectorate and ERPC, defining it as a state agency within the Ministry of the Environment whose main functions are provided for in the Act. This means that from 2004 Ministry of the Environment was responsible for issuing the licence and ERPC is responsible for preparing the safety review of the provided documents together with making the proposal to the Ministry. There was also need to clarify several requirements in the licensing procedure, and for example in 2006 the there were several amendments made to the legislation - providing the ERPC the right to question the licensee directly for additional information. In the Act, requirements were more precisely defined -

for example, there was one long article, which lists the responsibilities of the radiation practice licence owner. To obtain a radiation practice licence, an applicant shall submit an application and the following additional information to the Ministry of the Environment:

- the objective and description of the radiation practice,
- the layout of the location and facility for the radiation practice, and information concerning the technology and equipment to be used;
- the justification for and description of the radiation practice;
- information on the radiation source;
- information on the radioactive waste or emissions created in the process of the radiation practice, and concerning the radioactive waste storage facility waste and acceptance criteria thereof;
- the plan for rendering the radiation source harmless after termination of the use of the radiation source which, in the case of radiation practice involving moderate or high risk, must be approved by a qualified expert;
- a plan for radiation monitoring and information on the equipment to be used for radiation monitoring;
- radiation safety assessment and measures for guaranteeing radiation safety;
- an emergency response plan in the case of radiation practices involving high risk; description of the radiation safety quality system; information on exposed workers and their professional training.

Licenses are still provided for a period of 5 years. The Radiation Act states that a radiation practice licence shall set out the following: issuer, owner, the name of the radiation practice; validity of the radiation practice licence; a description of the radiation sources; the location where the radiation practice takes place and a description of the facility and premises; the manners in which radioactive waste is handled, and the maximum quantities and handling facilities for radioactive waste; the maximum quantities of radioactive emissions, and means of releasing them into the environment; the requirements for radiation safety and radiation monitoring arising from the given radiation practice and its specific character; and the risk category of the radiation practice. Proceedings arising from consideration of an application may take up to 90 days, which can be prolonged in the case of complicated activities for another 90 days. There is fixed state fee (~130 EUR) for proceedings, which goes directly to the state budget. The fee for proceedings does not depend on either the practice nor facility or equipment. There are no additional fees like a fixed annual cost for different kinds of radiation practices.

Radiation practices are divided into 3 risk categories depending on the risk presented by the radiation practice or the radiation source:

- low risk radiation practices, through which an exposed worker receives or is liable to receive an effective dose of up to 1 mSv in a year;
- moderate risk radiation practices, through which an exposed worker receives or is liable to receive an effective dose of up to 6 mSv in a year;
- high risk radiation practices, through which an exposed worker receives or is liable to receive an effective dose exceeding 6 mSv in a year.

High risk practices are for example radioactive waste management, activities related to nuclear fule cycle, HASS users (oncology, blood irradiation, industrial irradiation, industrial gamma-radiography), nuclear medicine, accelerator users; moderate risk practices are hospitals using X-ray diagnostic and treatment equipment. Industrial radiography using X-ray equipment is classified depend on

The following factors caused the 2004 changes in the overall system: according to the Estonian legislative base, the state agency has no right to issue the licences. So the only possibility of keeping the licensing in ERPC would have been changing its status to a governmental agency. However, it was not seriously discussed due to political reasons – a lack of political will to increase the number of state workers. It was also intended to keep licensing independent from the service provider. For example, the ERPC provides personal dosimetry service for the licensees. This artificial division has not really worked, as most of the communication with licensees is done by ERPC, and the licensees continue to consider the ERPC their licence provider. Until very recently, the Ministry of the Environment had been unable to build up the required competence in the field, and staffed with only a single specialist dealing with radiation protection questions. This has created a situation of continued reliance on the proposals made by ERPC.

According to the Radiation Act, the Environmental Inspectorate was to undertake inspections while ERPC provided the operational elements of the licensing function. The inspection plans were prepared in co-operation between the Environmental Inspectorate and ERPC. Risk levels of the radiation practice licence were taken into account when preparing the inspection plan: low risk practices were inspected in average once in 5 years, moderate risk practices once in 2-3 years and high risk practices each year. Since 2005 the Environmental Inspectorate and the ERPC have made inspections together, which has been good practice for both sides. The main issues that have arisen during inspection were that for example the licencee was unaware that a licence had already expired, or they did not inform in a timely manner regarding the use of additional radiation equipment. Those were also the main reasons for penalties.

Participation in the inspection process provides opportunities to validate licences in order to improve their quality in future. Much of the recent work of the ERPC has focused on licence reissuing. Good quality control processes have been developed, as the ERPC intends to increase its practice assessment activities to supplement its administrative approach. This will further ensure good radiation safety and security. Some of the most relevant learning points for the ERPC have been: that there is no need to provide the entirety of the Radiation Act on the licence because it will result in a long document which is difficult to read for the licensee and does not provide a cohesive source of relevant information in a short and concise manner. General conditions as set out in licence must be very clearly defined and explained. It has proved to be a useful practice to discuss possible special conditions with the applicant, to provide for a more realistic document. To ease the work and to make licensing more effective ERPC has prepared several education resources regarding how to fulfill the application and the specific kinds of data to present in the licensing procedure. This has resulted in a more harmonized approach used for licensing by the staff members of the ERPC and several procedures have been documented. It is an unwritten rule that all sites are visited before making the proposal to the Ministry.

The ERPC is responsible for maintaining several registries, for example for radiation licenses, sources, exposed workers. Since 2004 Ministry of Environment has issued about 700 licenses. On average, it has meant 100 licensing procedures per year, of which about 25 are new practices. Most of the licenses, around 60 %, are for dental x-ray equipment use, which means that a lot of time and energy has been devoted to dental application proceedings. One possible development foreseen is to move dental X-ray equipment to the notification category.

Over the last 2 years, a working group composed of representatives from different government ministries and other interested parties prepared a draft radiation protection action plan. This document is intended to cover primary actions for next decade. The preparation is now in the final stage - the environmental impact strategic assessment is done, most of the ministries have provided their comments to the document. It is expected that the draft legislation will be granted government approval in 2008. The action plan sets out the primary tasks of radiation protection in Estonia stating 4 main objective and actions to fulfil them. Sections covered include radioactive waste management, medical exposure, emergency preparedness, etc. One of the objectives is to alert or correct infrastructure deficiencies. It is designed to make easier and more understandable the proceedings of applications. At the present time the process includes 3 institutions. The fact that, first, the ERPC does not actually sign the licences, second that inspection is carried out by another body within the Ministry and third that the ERPC provides advice and services, could lead to significant confusion amongst users, the Ministry itself and potentially even within the ERPC. Further amendments to the Radiation Act are foreseen for 2009, which may also help to sort out some of the bureaucratic difficulties.

In Estonia, the first radiation practice licenses were granted a decade ago following Parliamentary approval of the first Radiation Act. Over the last ten years the concept of licence, as well as licensing system have changed remarkably. From a single page certificate to a lengthy and considered document with specialty requirements. There have been some continuing difficulties in defining the responsibilities of institutions involved. However, the licencing procedure is overall more easily understandable vis a vis the requirements as defined in the legislation. A very important component has also been preparation of different procedures and reccomendations. These growing pains have contributed significantly to providing a well developed overall picture of the Estonian situation and the development of appropriate radiation safety measures in Estonia.

Aktuellt från STUK om nytt kärnkraft i Finland

Olli Vilkamo

STUK - Strålsäkerhetscentralen

Informasjon om kärnkraft i Finland, bl.a. övervakning samt bygget av OL 3 reaktoren, miljökonsekvensbedömning av OL 4, Lo 3 och Fv 1 initiativen.

Issues for Neutronics Calculations for ITER Fusion Reactor

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Abstract

Risø DTU is developing neutronic calculations capabilities for magnetic thermonuclear fusion reactor technology. Requirements to neutron transport and activation calculations are addressed and an example of nuclear heating of the confining magnets of the ITER fusions reactor caused by proposed changes to the diagnostics system is presented.

Introduction

Neutronic calculation methods developed for fission reactor technology may readily be adopted to thermonuclear fusion reactor design studies, such as the ITER magnetic fusion reactor. Central neutronics issues in fusion reactor technology include power generation, shielding capacities, structural damage issues and the minimization of radioactive waste. In addition, there are issues specific to fusion power reactors such as tritium breeding and maintenance during reactor shutdown.

Neutronic calculations are required to ensure that existing or modified structures or components comply with ITER functional and technical requirements, and are used to [1]

- demonstrate capability of safe operation
- maintain, repair, replace or modify an item
- determine the cause of a malfunction of an item
- provide required baseline for access, repository and waste of items.

Since maintenance inside the cryostat by means of hands-on-operation may be necessary during shutdown and considering the minimization of radioactive waste it is important to minimize neutron activation in the cryostat, both with respect to short and long-lived isotopes. Thus the bulk radiation shield including the steel/water blanket, the vacuum vessel and other in-vessel and out-vessel components shall provide sufficient nuclear shielding, especially inside the cryostat at the port plug access areas. The ITER facility requires nuclear licensing because of its expected tritium inventory and the radioactive waste generation.

In the following a short introduction to the ITER is given and issues of fusion reactor neutron transport and activation calculations are addressed. Risø DTU is presently developing neutronic calculation capabilities for ITER, based on its experience within fission reactor technology. As an example of neutron and photon transport in the ITER environment calculations of nuclear heating of the superconducting magnets caused by proposed changes to the shielding blanket is presented.

ITER fusion reactor

The ITER thermonuclear fusion reactor is designed to be the next step towards developing a commercial fusion power plant. In the tokamak layout of ITER, strong magnetic fields confine a deuterium-tritium plasma, in which prolonged fusion power production takes place.

ITER is being constructed at Cadarache in the south of France. The fusion reactor is expected to start operating in 2016 and construction costs are estimated at 4.5 billion Euro. The development, construction and operation of ITER are a joint undertaking by the European Union, China, India, Japan, South Korea, the Russian Federation and USA. European participation is coordinated by EFDA, the European Fusion Development Agreement, which coordinate all European activities within the field of fusion research of magnetic confinement. It is managed by the EURATOM Associations and the European Commission.

The overall goal of ITER is to demonstrate the scientific and technical feasibility of fusion power. More specifically, the Q value, representing the amount of thermal energy that is generated by the fusion reactions, divided by the amount of external heating, should reach a peak value of 10, and a value of Q=5 should be sustained for prolonged periods, allowing for a "burning plasma". This is to be compared with JET, presently the largest tokamak in the world, which has reached Q=0.65, near the point of "break even" (Q=1). At Q=5, most of the heating of the plasma comes from the fusion reactions themselves.

Secondary goals of ITER are to implement and test technologies and processes needed for future fusion power plants, including superconducting magnets and remote handling, and to test and develop concepts for tritium breeding.

A schematic layout of ITER is shown in Figure 1.

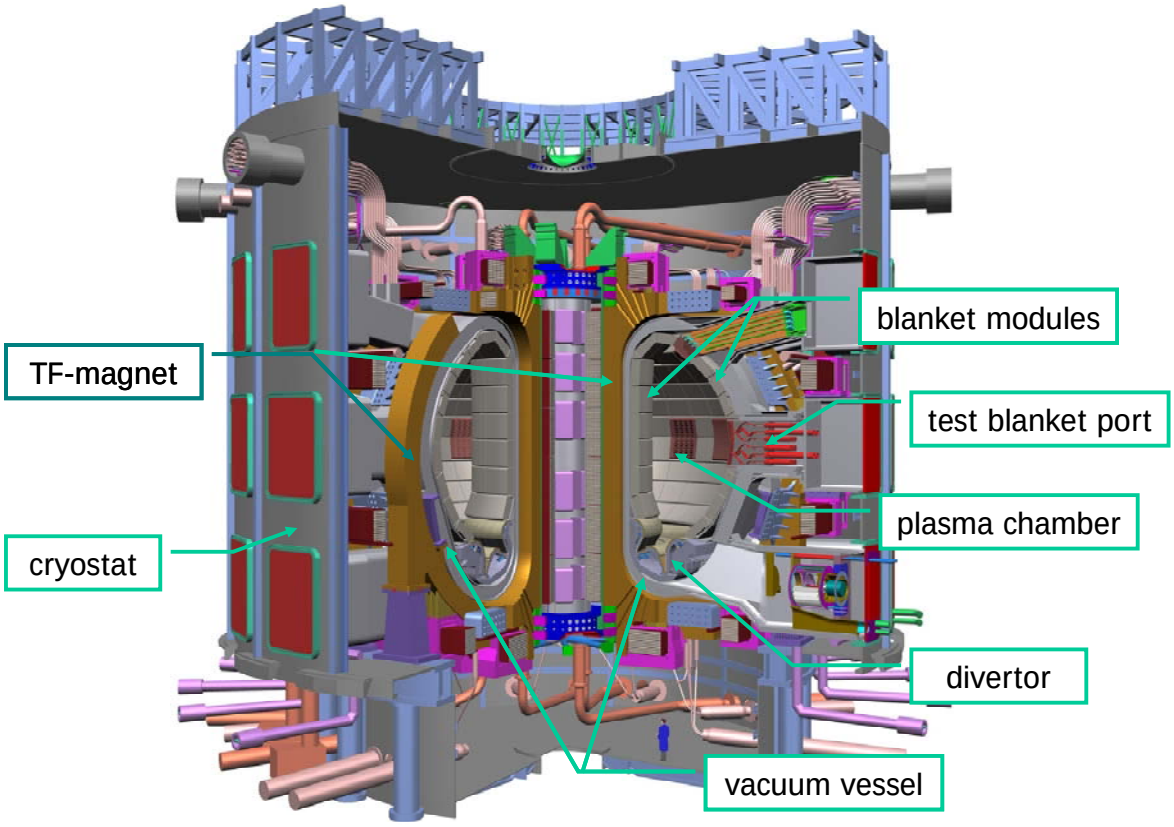

Figure 1. *Schematic layout of the ITER fusion reactor.*

Height=24 m, plasma volume=850 m^3, power=500 MW.

Risø DTU participates in the European coordinated programme for fusion research in close collaboration with other European and international groups. The two main activities are modelling of turbulence and transport in the fusion plasma and development of a Collective Thomson Scattering (CTS) diagnostics for measuring fast ions in the plasma. Part of the CTS system is located inside the blanket where the fast neutron flux is very high and neutronic calculation is a key element in the detailed designing of the system.

Neutronic calculations for fission and fusion reactors

Neutron transport calculations are carried out either using deterministic models or by Monte Carlo calculation. A main difference in neutronic calculations between the fission and fusion reactors is the neutron source term: While in fission reactors the neutron flux is both generated by, and determines the fission reactions, which must be solved by criticality calculations, the fusion reactions do not depend on the neutron flux, but are determined solely by the plasma conditions. Hence, in neutronics calculations for fusion reactors the source term is externally prescribed, dictated by the design specifications of the reactor.

A second difference between fission and fusion reactors is the neutron energies. The DT fusion process,

$$\mathrm{D + T \rightarrow {}^4He + n + 17.6\ MeV,}$$

generates neutrons with kinetic energy 14.1 MeV (and alphas of 3.5 MeV). A comparison between the energy spectrum of fusion neutrons and the fission neutrons is shown in Figure 2. It is seen that while the fission spectrum peaks around 1–2 MeV the fusion neutron spectrum at the first wall is dominated by the 14 MeV primary neutrons. For Monte Carlo transport calculations this implies that cross section libraries for fission reactors must be extended to cover fusion neutron energies.

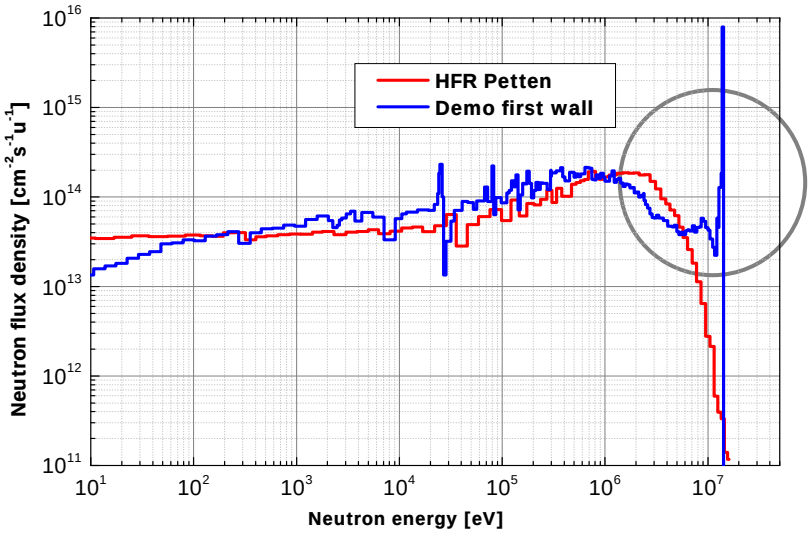

Figure 2. *Neutron spectra in fission and fusion reactors [2].*

Specific to thermonuclear fusion reactor neutronics is tritium breeding calculations. The fuel for the DT fusion reaction consists of water (containing deuterium) and lithium. Tritium breeding is obtained

through one of the two following processes, where the first involves thermal neutrons and the last fast neutrons (>2.5 MeV),

$$^{6}Li + n \rightarrow T + {}^{4}He + 4.86\ MeV$$

$$^{7}Li + n \rightarrow T + {}^{4}He + n - 2.5\ MeV.$$

To ensure that a sufficient number of neutrons are available for tritium breeding, neutrons are generated through multiplication, in part stemming from the above reaction and in part from interactions of fast neutrons with the first wall containing beryllium,

$$^{9}Be + n \rightarrow 2n + 2\ {}^{4}He.$$

With respect to the ITER design, general neutronic calculation issues are

Blanket: Tritium breeding, power generation

- Assure tritium self-sufficiency, provide nuclear heating data for thermal-hydraulic layout

Shield: Attenuate radiation to tolerable level

- Assure sufficient protection of super-conducting magnets
- Avoid structural damage, helium gas production in steel structure

Safety & Environment, Maintenance: Material activation

- Minimise activation inventory with regard to short-term and long-term hazard potential
- Maintenance service during reactor shutdown (dose level)

Requirement to neutron transport and activation calculations

A quality assurance program has been established for ITER activities, including neutronic calculations [3]. It stipulates that such calculations shall

- conform to established requirements
- be fully traceable
- be capable of withstanding detailed technical review.

Neutronics calculations are central to design, safety and ultimately nuclear performance of a system. The calculations are used to size the shield, determine components lifetimes, estimate damage levels and assess the radiation environment of the reactor.

Quality assurance and quality control in neutronics calculations are obtained through using only qualified computational simulations, hereunder appropriate computational methods, tools (codes) and data (nuclear cross-sections), validation procedures, and from qualification through integral benchmark experiments.

Monte Carlo neutronic calculations for ITER diagnostics

The proposed CTS diagnostic for ITER consists of two separate systems measuring the fast ion velocity components in directions near parallel and near perpendicular to the magnetic field [3]. One of these systems requires alterations to an inboard blanket module in the form of a cavity and a slot to accommodate a CTS receiver unit. The cut-off of the blanket implies increased neutron and gamma fluxes to the magnetic system, in particular to two of the inboard toroidal field (TF) coil legs.

Monte Carlo calculations of neutron and gamma fluxes resulting from the proposed modification of the blanket module have been carried out using the MCNP-4C code. To this purpose, a geometrical model of the tokamak system including blanket, vacuum vessel and TF coils was developed at Risø DTU. The model is a simplification of the ITER-FEAT model applied in 3D neutronics analyses of ITER components and allows for 3D parametric studies.

The inboard part of simplified model is shown in Figure 3. The CTS receiver is placed inside a cavity on the back side of the blanket module (mirror cavity) and requires a horizontal slot (cut-off) facing the plasma. The vertical gap of the slot should be no less than 30 mm in order to satisfy the measurement requirements.

In Figure 4 the effect of the design modifications on the neutron and gamma heating of the TF coil is shown. The cavity gives rise to an increase in the nuclear heating of approx. 50 % while inserting a 30 mm slot increases the nuclear heating by additional 50%. The nuclear heating of the TF coil remains below the design limit, which is 1.4×10^{-4} Wg^{-1} [4].

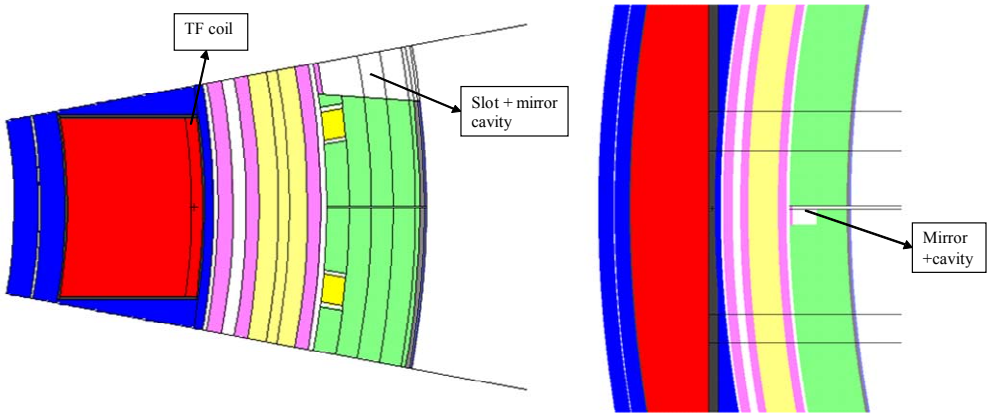

Figure 3. *Horizontal (left) and vertical (right) cross sections showing the CTS mirror cavity and slot.*

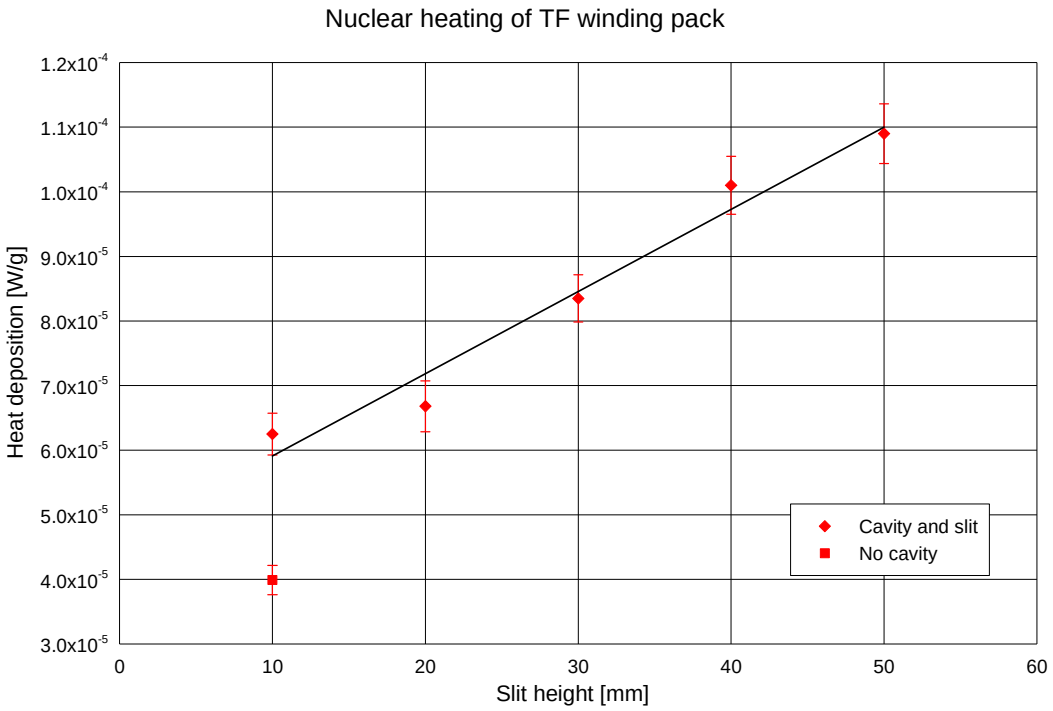

Figure 4. *Local nuclear heating of the TF winding pack.*

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