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RADIATION PROTECTION IN THE 2000S – THEORY AND PRACTICE

Nordic Society for Radiation Protection

Proceedings of the XIII ordinary meeting

Turku/Åbo, Finland
August 25 - 29, 2002

W. Paile (Ed.)

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Introduction

Radiation protection is a multidisciplinary field, the different sectors of which have deep impact upon each other. It is therefore important for anyone working with radiation protection to keep up a certain level of insight into all areas of the field, in addition to being an expert in the own working area. For this purpose, a conference gathering together participants from the whole field of radiation protection has a valuable role to fulfil.

Nordic cooperation has long traditions in radiation protection, which is natural considering the common way of thinking when it comes to matters of security. As a result, Nordic regulations have been quite well harmonized long before the implementation of international safety standards. The heads of the Nordic radiation protection authorities meet regularly. There is also a lot of scientific cooperation within the frames of different projects.

The Nordic Society for Radiation Protection, NSFS, is an associate society of the International Radiation Protection Association IRPA. The members of the society are professionals representing all different areas related to radiation protection. Ordinary meetings are held every 3 year. The XIII ordinary meeting was held in Turku/Åbo, Finland, in 2002. 160 professionals took part in the meeting. The participants came from all the Nordic countries and also from Bulgaria, Estonia, and Poland.

Separate sessions were held on protection of the environment, information and ethics, natural radiation, reactor safety and waste management, emergency preparedness, medical use of radiation, radiation in industrial use, education and certification, radiation biology and epidemiology, radioecology and monitoring. An introductory lecture was given by the recipient of the Bo Lindell-award, Professor Sören Mattsson. In all, there were 57 oral presentations and 44 poster presentations. The greatest number of papers were presented in the sessions on medical use (21 papers), radioecology and monitoring (20 papers), and natural radiation (13 papers). For these proceedings, all papers presenting scientific results have been subject to a referee process with evaluation of independent experts.

I hereby want to express my thanks to all those presenting papers during the conference and also to all the other participants. They all contributed to make the conference a success. It is my sincere feeling that the conference, bringing together people from different fields, well fulfilled the purpose of broadening the professional views of the participants.

Helsinki, May 2003 Wendla Paile

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Avainsanat NSFS, pohjoimainen, säteily

Esipuhe

Säteilysuojelu on poikkitieteellinen ala, jonka eri osa-alueet vaikuttavat voimakkaasti toisiinsa. Tämän vuoksi on tärkeää, että jokainen säteilysuojelun parissa työskentelevä pysyy riittävässä määrin ajan tasalla sen kaikilla osa-aloilla, sen lisäksi, että toimii oman alansa asiantuntijana. Konferenssi, joka kokoaa yhteen osanottajia säteilysuojelun kaikilta aloilta, voi arvokkaalla tavalla palvella tätä tarkoitusta.

Pohjoismaisella yhteistyöllä on pitkät perinteet säteilysuojelussa. Tämä on varsin luonnollista, kun ottaa huomioon yhteistä turvallisuuskulttuuriamme. Pohjoismaiset säännöstöt ovatkin olleet hyvin keskenään harmonisoituja jo kauan ennen kuin kansainväliset säteilysuojelustandardit on otettu käyttöön. Pohjoismaisten säteilysuojeluviranomaisten johtajat tapaavat myös säännöllisesti toisiaan. Lisäksi tehdään paljon tieteellistä yhteistyötä erilaisten projektien muodossa.

Pohjoismainen säteilysuojeluseura NSFS (Nordiska sällskapet för strålskydd) on kansainvälisen säteilysuojeluyhdistyksen IRPA:n jäsenyhdistys. Seuran jäsenet ovat säteilysuojelun ammattilaisia ja edustavat kaikkia säteilysuojelun osa-alueita. Seura järjestää yleiskokouksen joka kolmas vuosi. Kolmastoista yleiskokous pidettiin Turussa vuonna 2002. Tuolloin 160 alan ammattilaista osallistui kokoukseen ja osanottajia saapui kaikista pohjoismaista sekä myös Bulgariasta, Virosta ja Puolasta.

Erilliset sessiot järjestettiin seuraavilta aloilta: ympäristönsuojelu, viestintä ja etiikka, luonnonsäteily, reaktoriturvallisuus ja jätehuolto, valmius, säteilyn lääketieteellinen käyttö, säteilyn käyttö teollisuudessa, koulutus ja sertifiointi, säteilybiologia ja epidemiologia sekä radioekologia ja mittausmenetelmät. Konferenssin avausluennon piti Bo Lindell-palkinnon saaja, professori Sören Mattsson. Kaiken kaikkiaan konferenssissa kuultiin 57 suullista esitystä ja nähtiin 44 posteria. Eniten esityksiä kertyi seuraaviin sessioihin: säteilyn lääketieteellinen käyttö (21 esitystä), radioekologia ja mittausmenetelmät (20 esitystä) ja luonnonsäteily (13 esitystä). Tätä kokousjulkaisua varten kaikki esitykset, joissa on esitetty tieteellisiä tutkimustuloksia, ovat käyneet läpi referee-tarkistuksen, jossa riippumattomat asiantuntijat ovat arvioineet tekstit.

Haluan tässä esittää kiitokseni kaikille niille, jotka osallistuivat konferenssiin tieteellisellä esityksellä sekä myös kaikkia muita osallistujia. Kaikki te olette osaltanne myötävaikuttaneet onnistuneen kokouksen toteutumiseen. Uskon vilpittömästi, että tämä kokous on täyttänyt hyvin tehtävänsä tuodessaan yhteen eri alojen ihmisiä, täten laajentaen osanottajien ammatillista näkemystä.

Helsingissä, toukokuussa 2003 Wendla Paile

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SESSION 1. OPENING OF CONFERENCE

The Bo Lindell Lecture. Radiation protection among populations, patients, personnel, plankton and plants – What can we learn from one another?

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I sincerely would like to thank the Nordic Society for Radiation Protection for inviting me to present the Bo Lindell lecture for 2002. It is a great honour and privilege to do that. We are many friends of Bo who are grateful to him for his objectivity and stringency and for always succeeding to combine his leading role in national and international radiation protection with generous support and inspiration to colleagues.

Introduction

The field of radiation protection is broad. It ranges from the management of radiation and radioactive material at many different places of work, the use of radiation and radiopharmaceuticals for medical investigations and treatments and the radiation exposure of ourselves and our environment from natural and man-made sources to the current discussion on how to manage low-level exposures. No single person can be expert in all these fields. They have however many things in common: basic radiobiology, radiation detection and dosimetry, risk considerations and need to inform other people. It might also be helpful when we form our own views on radiation protection if we can refer to other areas than our own. In this lecture, I will go through various fields of radiation protection and give some personal comments. I will also try to identify some matters of importance for the future work.

Human populations

The protection of man will continue to dominate our efforts with a priority for the youngest individuals (including embryo and foetuses), which we believe are the most sensitive to radiation. Today we are around 6 000 million people in the world. Our natural exposure gives an effective dose of 2.4 mSv per capita, the main sources being cosmic radiation, terrestrial gamma rays, ingestion of naturally occurring radionuclides (^{40}K) and radon daughters in indoor air (UNSCEAR, 2000). Seen on a global scale, we can hardly change that value very much, just locally try to reduce the highest exposures. With increasing standard of living, man-made radiation sources such as radiation in medicine, releases from nuclear industries, high-level waste, consumer products and accidents will play an increasing role. All of us, rich or poor, do still have reasons to be afraid of the military use of nuclear weapons. The risk that non-national groups or individuals use primitive nuclear weapons or “dirty bombs” has become real and we can not neglect the risk for new large reactor accidents. It is necessary to continue the efforts to improve our preparedness for all these situations.

Patients

Medical procedures represent the largest source of human-made radiation exposures. There are 2 000 million medical x-ray investigations and 300 million dental x-ray investigations made each year in the world. Most of them (3/4 and 9/10 respectively) are carried out on 1/4 of the world’s population (for which the medical investigations give an effective dose per capita of 1.2 mSv/year). On top of that there are 32 million investigations in diagnostic nuclear medicine and 5.5 million treatments per year in radiation therapy (UNSCEAR, 2000; Shrimpton, 2001). In our part of the world, there has been a silent but dramatic change of CT-investigations (more procedures, more scans per procedure). Today CT accounts for 5% of the investigations but gives rise to 50(!)% of the collective dose from diagnostic radiology. Interventional procedures, increasingly being used to replace surgery, can lead to very high local skin doses, in some cases exceeding the threshold for deterministic effects (ICRP, 2000). For most X-ray examinations, there are still too large inter-hospital variations in doses for the same type of examination. Digital radiology, which has the potential to lower the doses, for most investigations still shows higher doses than conventional film-screen technique. On the other hand, the development in mammography (in many countries used for repeated breast cancer screening) shows how real improvements can be done. The average glandular dose has been reduced from 100 mGy (3 projections) in the 1970´s with direct film to 1 mGy as a standard today (vacuum cassette, no grid) and a potential for 0.2-0.3 mGy with the most recent digital technique (Hemdal et al., 2002).

Personnel

The care for the personnel in radiology was the start of the radiation protection. Today there are 2.7 million occupationally exposed persons of which 3/4 works in the health care sector. The doses are very low in diagnostic radiology, somewhat higher in nuclear medicine and still somewhat higher at nuclear power stations. For personal dosimetry, there are some remaining dosimetric problems related to the weighting of the neutron components in reactor environments, at waste handling and at high-energy accelerators for radiotherapy as well as for physics research.

"Plankton and plants" - organisms other than man

Up to now the radiation protection criteria have been based on minimising the risk to humans. Today, the efforts are closely related to the discussion about a sustainable environment which most of us welcome. It s a challenge to obtain tools to assess and evaluate long-term consequences of radiation exposure to the whole biosphere. Today much work is going on to define a limited set of reference organisms with their relevant reference dose models and data sets (Pentreath and Woodhead, 2001). We have to realise that this is a real long-term project, with many parallels to the development of a "standard man", standard human biokinetic models and dose catalogues for man during the last 30 years.

What do we have in common and what can we learn from one another?

There are basic knowledge which is common for all branches of radiation protection:

Radiobiology: The rapid development of cellular biochemistry and molecular biology will have great impact on the understanding of the action of radiation on cells, of DNA damage, repair processes and dose rate effects. It is important that this information is translated into the theory and practice of radiation protection as soon as there is consensus about the new data.

Epidemiology: The current knowledge of health effects of ionising radiation on populations, including the base for our current radiation protection philosophy, the linear non-threshold hypothesis, is supported by a number of human epidemiological studies. The most important is still the follow up of the atomic bomb survivors at Hiroshima and Nagasaki. Great efforts have also been made to recognise excess leukaemia and solid cancers among groups of patients and nuclear workers. For the latter groups, the doses to the individuals have normally been too low and/or the groups too small to provide significantly improved risk estimates. These studies have only demonstrated that radiation risks can not greatly exceed current ICRP-estimates. Apart from the new information on thyroid cancers in children due to radioiodine, this statement also seems to be valid for the Chernobyl accident. There is now another major source of information on the health effects of human exposure emerging in the Southern Urals, from the follow up of the nuclear workers at the plutonium production facility Mayak and of the people who were exposed to fission products from Mayak that contaminated the Techa river (Kellerer, 2002). Also other former Soviet facilities might give us new independent risk data.

Radiation physics and dosimetry will continue to be critical as will the **detection methods:** For the development of criteria for other species than man, the development of a reliable dosimetry is as important as that for man and an important cornerstone in the future course of progress in the field. It is straight forward to extend the current MIRD dosimetry concept and other types of Monte Carlo calculations to volumes of interest for other species than man. For man, improved voxel phantoms useful for patients as well as for occupationally exposed persons and for members of the general public need to be developed. Much research related to diagnostic and therapeutic nuclear medicine (e.g. biokinetic and metabolic studies) are also of interest for occupational exposure and exposure of the general public and vice versa. Development of new detectors and dosemeters will simplify experimental research (amorphous silicon, new material for thermoluminescence and optically stimulated luminescence dosimetry).

Effective dose, risk estimates and comparisons: The effective dose concept was derived for occupationally exposed workers and was from the beginning also intended for the general public. The concept has also been found to be very practical for patients as medical exposures often are limited to specific body parts or are very inhomogeneous in distribution. The concept of effective dose has been an important instrument to get a dialogue about radiation exposure and risks in medicine and between various areas of radiation protection. Especially in medicine, there is a need to develop a weighted risk factor or a new way to calculate collective dose contributions, as the typical patient is old and often also sick, and therefore has less life expectancy during which late effects may occur than the average member of the public.

The base and reference value for all our comparisons is the natural background. The inclusion of the radon daughter component has introduced confusion and continuos alterations of this basic reference value. I propose that we again leave the radon daughter contribution to lungs and airways out and use 1 mSv per year from cosmic radiation, natural potassium in the body and external gamma radiation as a basis for our judgements on risk increments from exposure from man-made sources.

Collective dose: The use of the concept collective dose has been debated heavily during recent years. It is a logical consequence of the LNT-model (Lindell, 2001) and is a sound instrument for planning and making priorities for patients as well as for personnel.

Overall policy for radiation and other risk factors: It is of course important that the criteria for ionising radiation are based on the same overall policy as the criteria for other potentially harmful agents (non-ionising radiation, chemicals, CO_2, etc) which may affect man and nature. It is well-known that radiation protection for long has been an important forerunner in formulating safety criteria. Within medicine we need to widen our risk discussion, not just to concentrate on radiation risks. We have for instance to include risks from X-ray contrast agents and the risk to die in an MR-investigation due to flying metal objects. We should also quantify the risk not to do an investigation. A problem lies in the fact that the assessment of benefit seems to be even more difficult than estimating radiation risks.

Use of existing information: Research in environmental radiation protection has contributed to the understanding of global transport processes in stratosphere, troposphere and atmosphere as well as in the marine environment. This is information which also can be used to tell us much about the transport of chlorinated hydrocarbons, heavy metal, other toxic substances and nutrients. Much of the existing radiobiology is based on animal experiments - information which could be of direct use for the protection of other species than man. It is also important that we today take advantage of the old studies related to environmental effects of radiation. One example is the investigations of effects of radiation on an ecosystem level carried out during the cold war (Lidén, 1972) to study the radiation effects of nuclear weapons. The research related to radiation doses and effects on man has often also included the transfer of radionuclides in the environment and sometimes also doses to other species than man. It is therefore likely that significant harm to population of other species would already have been detected (Salo, 1999). Bioindicators are now again of interest because they have high activity concentrations and therefore can be expected to be the first species to show any effects. Attempts were done from time to time to calculate the absorbed dose to these indicator organisms and to animals which were eating them. New studies of bioindicators could hopefully be combined with studies of markers for radiation effects developed in molecular biology so that dose/effect information can be obtained. A "problem" is however that in most situations the contamination is so low that the absorbed doses from the naturally occurring radionuclides as ^{40}K and ^{210}Po totally dominate over the contributions from the artificial radionuclides (such as ^{137}Cs, ^{60}Co, ^{99}Tc and $^{239,240}Pu$). So again we perhaps need large scale experimental studies.

High-level waste and other radioactive sources: The safe disposal of our spent nuclear fuel and other high-level waste will be a matter of continued interest and importance for the future and acceptable disposal sites have to be selected. This is a question which also has an ethical and a social dimension and is much connected to the question of information and acceptance. There are also other radioactive sources which need to be discussed. There have been 400 significant accidents involving sources of radiation: 3 000 people have been exposed, of whom 100 have died from acute effects. Half of these fatalities have been a result of accidents in medicine. Over 500 000 sources have been produced. Of these about 100 000 are still in use (Renata Czarwinski of the German Bundesamt für Strahlenschutz cited by Kendall, 2001). We have to keep track of these sources and simplify their safe disposal.

Education and training: When discussing radiation protection in medicine, we see a need for education of radiologists and referring physicians. It is obvious that a specialist in diagnostic radiology should be familiar with basic radiation protection and that is also the view of their own professional organisations. However, this is not the case in all our Nordic countries.

Information and acceptance: We have to communicate the scientific and technical facts to the public in an understandable way. We must never have an attitude that this is so complicated that people can not understand it. Information and participation is requested by individuals and local groups including representatives for the communities which are candidates for disposal of nuclear waste. Also patients expect to be adequately informed about the radiation exposure connected to a proposed investigation or treatment, about the benefits and risks, about the alternatives and about the duties of the medical staff.

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ICRP and the Progress Towards New Recommendations

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Abstract

The International Commission on Radiological Protection (ICRP) is currently revising its recommen-dations and is considering a simpler individual-oriented approach. The major changes will be an emphasis on the protection of individuals; broadening of the concept of dose limits by adding a range of protective actions and levels above which action should be taken; a distinction between protective actions that can be applied to the source and those that can be applied to the pathways from the source to the individuals; clarification of the dosimetric quantities; and a framework for protection of non-human organisms. For the revision, ICRP has set up a Task Group, consisting of the chairman, vice-chairman, and the chairman of each of the Committees. This article presents some of the proposed changes in the Commission's recommendations for the start of the 21st century.

Introduction

ICRP's advice is aimed principally at the regulators and managements that have direct responsibility for establishing protection standards. The present recommendations of ICRP were developed over some 30 years and published in 1991 [1]. The system has become increasingly complex, and it has in some respects been difficult to explain the variations between different applications. The Commission intends to simplify its system of protection and make it more transparent [2]. The Commission will propose the following major changes:

- an emphasis on the protection of individuals;
- broadening of the concept of dose limits by adding a range of protective actions and levels above which action should be taken;
- a distinction between protective actions that can be applied to the source and those that can be applied to the pathways leading from the source to the doses in individuals;
- clarification of the dosimetric quantities; and
- a framework for protection of non-human organisms.

Underlying policies

There are several underlying policies for the revision of ICRP's recommendations. Where exposures can be avoided or controlled, there is a duty to provide appropriate mini-mum levels of protection for the exposed individuals and for society as a whole. Because there is some risk even from small radiation exposures, there is a further duty to take steps to provide higher levels of pro-tection. The global average annual effective dose from natural sources is about 2.4 mSv [3], and radiological protection concerns the incremental doses due to additional sources, many of which cannot be avoided or controlled. In practice, protection is concerned with the risks at levels above a few millisieverts in a year. The system of protection is concerned mainly with the management of stochastic effects, as the total doses from all sources for most individuals are well below the level that might cause determinis-tic effects. At these dose levels, the probability of harm is presumed to be proportional to the dose. Each source can be considered separately, and no sing-le protective action can act on the total dose to an individual. For some individuals, the dose from one or two sources may be high enough to cause deterministic effects, and these sources will be dealt with outside the general system of protection.

The recommendations can be applied only to situations in which either the source of exposure or the pathways leading to the doses received by individuals can be controlled. ICRP calls such sources *controllable sources*. Individuals are exposed to both natural background radiation and to controllable sources. There are also sources for which the resulting effective dose are very low or for which the combination of dose and difficulty of applying controls are such that the Commission will exclude them from its recommendations.

The principle of justification: The responsi-bility for the justification of a practice usually falls on governments or government authorities. ICRP will apply its system of protection to practices that have been justified. Sources that can be controlled only by environmental action may have been introduced by previous practices. Because these sources already exist, justification is not relevant and they are therefore within the scope of the recommendations. In medicine, there are three levels of justification: (1) the use of radiation is accepted as doing more good than harm, and its justification is now taken for granted; (2) a specified procedure with a specified objective is defined and justified; and (3) the application of the procedure to an individual patient should be justified.

Natural sources: ICRP has recommendations for protection against radon-222 in dwellings and workplaces [4], and the Commission will propose that the existing recommendations continue because they have wide acceptance. The regulatory authorities are expected to set Action Levels below which exposures are excluded from the system of protection. The Exposures to cosmic rays at ground level are not controllable, and they will be excluded from the scope of the recommendations. ICRP has previously recommended that the exposures of aircrew should be treated as occupational exposure in its system of protection, and there is no justification for including doses to members of the public.

Medical sources: Medical uses of radiation for patients require separate guidance, because limitation of the dose to the individual patient is not recommended as it may reduce the effectiveness of the patient's diagnosis or treatment. Instead the emphasis is on the justification of the medical procedures, and the optimisation concentrates on the requirement to keep the doses to patients as low as is consistent with the medical objectives. Diagnostic Reference Levels are used to indicate that, in routine conditions, the dose to the patient from a specified diagnostic procedure should not normally exceed the reference level for that procedure.

Dosimetric quantities: Effective dose will be the principal quantity for use in radiological protection. For the purpose of managing a radiation injury in an individual, estimates of absorbed dose, possibly weighted by the radiation-weighting factor, will be needed. Radiation-weighting factors are deter-mined by the characteristics of the type and energy of the radiation. The existing tissue-weighting factors are based on complex reasoning, and not solely on the fatality risk. The two types of weighting factors need some simplification and values have yet to be selected.

The revised system of protection

The revised system of protection will contain a structure of practical recommendations for protective actions associated with each relevant source. These actions will be aimed at providing a necessary *basic level of protection* for individuals, and the standard is set by a structure of conditions, **Protective Action Levels**, requiring specified protective actions. Optimisation of protection is needed to provide the *best level of protection* of individuals. In addition to the protection of individuals, the application of the optimisation of protection should be applied to groups. This system of protecting individuals and groups will be similar to the previous system, and will allow the protection to be applied with emphasis on the areas where protection is most needed. The Protective Action Levels will replace the old dose limits, intervention levels, action levels, constraints and exemption levels, and each protective action level will be associated with the action to be taken. Where available, protective actions applied at the source are to be preferred. ICRP will provide guidance about the protective action levels and the associated types of protective action and about the optimisation of protection. The guidance will be broadly consistent with the 1990 recommendations [1] but with a different emphasis.

The new Protective Action Levels can provide the basis for international agreements on restriction of dose or exposure. This is not sufficient and additional requirements are needed to achieve a sufficient standard of protection for individuals. ICRP will specify the combination of protective action and Protective Action Level for a single source and a representative individual. In most situations, there is more than one source and more than one type of individual. If doses are high enough to call for protective action, one source is usually found to be predominant. Each of the combinations can then be applied separately.

The dose used in the Protective Action Levels applies to an individual. There are at least three types of exposed individuals, namely patients; specially trained workers; and general workers and the public. The exposure of patients is usually voluntary and medical exposure of patients will be therefore dealt with separately. Workers in ‘controlled workplaces’ are specially trained. The general workers and of members of the public have little influence over their exposures and they should be treated as a separate group.

There are several factors that influence the choice of Protective Action Levels. The global average effective dose from natural sources excluding radon is about 1 mSv per year, and this dose can be a starting point for selecting protective action levels. The natural background provides no justification for additional exposures, but it can be used as a basis of judgement about the importance of other exposures. Additional doses of about 100 times this dose are likely to be a matter of high concern. Additional doses from justified practices far below the natural annual dose should not be of concern to the individual or to society. In the intermediate region, the doses are sometimes a legitimate matter for significant concern, calling for protective action. A general scheme for the degree of concern and the level of exposure as a fraction or multiple of the average natural background, excluding radon, is shown in Table 1.

ICRP will propose a general Protective Action Level of 100 mSv. At higher individual doses, the risk from a source cannot be justified, except in

Table 1. Levels of concern and individual effective dose in a year in relation to the average natural background, excluding radon.

Level of Concern	Effective Dose (mSv)
High	> 100 x
Raised	> 10 x
'Normal'	Natural Background (1-10)
Low	< 0.1 x
None	< 0.01 x

extraordinary situations, such as space flights or life-saving measures in accidents. No protective action will be called for if the effective dose is less than 0.01 mSv in a year. It is not possible to receive a dose of 0.01 mSv in isolation, and, therefore, the associated risk is for increments above background and to associate a risk with 0.01 mSv is meaningless. At higher levels of exposure, some tens of mSv, the total dose may be associated with a risk using the nominal risk coefficients. ICRP will give more advice on this issue. The recommended Protective Action Levels and the associated actions are shown in Table 2.

The optimisation of protection

Because of the presumption that there is some adverse health effect from exposures to ionising radiation even at small doses, ICRP's recommendations include a requirement to optimise protection for a given source. This requires the achievement of *the best level of protection* that is reasonable. Doses below about 0.01 mSv in a year, or less than one percent of the background dose excluding radon, from a source need not be regarded as significant for this purpose. The optimisation of protection applies to the protection of single individuals and also to the protection of groups. This optimisation procedure is essentially one of judge-ment and this process may best be carried out by involving all those most directly concerned, including operators, regulators, and representatives of those exposed, in determining or in negotiating the best level of protection. ICRP's recommendations need not deal with this degree of societal process.

In addition to the protection of individuals, there is an additional requirement to protect groups, taking account of the changes in both the level of dose and the number of individuals affected. No single property of this distribution is adequate for making these comparisons, and for most purposes

Table 2. Recommended protective action levels and the associated actions.

Exposed Group	Protective Action Level[1]	Associated Action
All situations		
All	< 0.01 mSv	Exclude from the system of protection
Prolonged situations		
Members of the public and general workers	0.1 mSv	Reduce doses
Radiation workers	20 mSv	Reduce doses
Single events and accidents		
Members of the public, general workers	5 mSv	Advise sheltering in buildings
	50 mSv	Arrange short-term evacuation
	1000 mSv (long-term)	Arrange long-term relocation
Radiation workers	1000 mSv	Upper level for planned emergency work

[1] The values are taken from ICRP Publications 60, 63, 64, 77 and 82.

several indicators will have to be used. For the protection of groups, the minimum combination of parameters would be the arithmetic mean dose and the number of individuals in the group. The product of these two parameters, the collective dose, is of limited utility because it aggregates the information excessively. This concept was introduced originally to facilitate cost-benefit analysis, and as a means of restricting the uncontrolled build-up of exposure to long-lived radionuclides in the environment from what was, at the time, imagined to be a global expansion of nuclear facilities. A large dose to a few people is not equivalent to a small dose to many people, and the collective dose should not be used in making decisions. To avoid excessive aggregation, it will often be helpful to present the necessary information in the form of a matrix.

Protection of the environment

ICRP has, up till now, not dealt with environmental protection, and there are therefore no ICRP recommendations as to how radiological protection of the environment should be carried out. The human habitat has probably been afforded a fairly high level of protection through the application of the current system for protecting humans. There is, however, no internationally endorsed system or criteria to demonstrate how, and to what extent, the environment is protected. Given the lack of a systematic and structured approach that has wide support, there are strong expectations from many quarters on the ICRP to act. ICRP has therefore set up a Task Group with the aim of developing a protection policy for, and suggesting a frame-work of, protection of non-human organisms.

The ICRP Task Group does not intend to define dose limits for biota. It will recommend a framework that can be used to provide high-level advice and guidance and help regulators and operators demonstrate compliance with existing environmental legislation. It is also necessary to develop an international system that is practical and that acknowledges the present level of knowledge concerning radiation effects on biota.

Although the Task Group has not yet finally decided on the objectives for the environment, these might be to *safeguard the environment* by preventing or reducing the frequency of effects likely to cause early mortality, reduced reproductive success, or the occurrence of observable cytogenetic effects in individual fauna and flora to a level where they would have a negligible impact on conservation of species, maintenance of biodiversity, or the health and status of natural habitats or communities. To achieve these objectives, a set of reference dose models, reference dose per unit intake and external exposure values will required, plus reference data sets of doses and effects for both man and the environment.

At its meeting in May, 2002, the Main Commission gave strong support to the work of the Task Group and emphasized the need to develop a system along the lines described above. The Task Group's report will be put on ICRP's website for consultation this year and will also be discussed at international meetings over the next two years.

Discussion

ICRP's system of protection has evolved over time as our understanding of underlying mechanisms has increased, and substantial revisions are made at intervals of about 10-15 years. The ideas of the ongoing revision have already been published in international journals and promulgated through the International Radiation Protection Association (IRPA). It is therefore likely that any system designed for the radiological protection of the environment would also take time to develop, and similarly be subject to revision as new information is obtained and experience gained in putting it into practice.

The Commission wishes to see an on-going debate over the next few years, and in 2004 the proposal for new recommendations on radiological protection will be discussed at the 11th IRPA conference in Madrid. The new set of Recommendations will be adopted by the Commission in 2005. This will be 15 years after the current recommendations were adopted.

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Epidemiological evidence of cancer risk - implication for low dose risk assessment and LNT concept

Per Hall

The major strength of epidemiological studies is that human beings are under study; risk is thus not extrapolated from animal data or derived from molecular studies. When interpreting the results it has to be kept in mind, however, that there are several weaknesses in epidemiological studies, especially studying low doses and their effects.

Epidemiological studies are observational rather than experimental in their design. This is particularly true for what is called etiological epidemiological studies. It would, for instance, probably be very hard convincing any ethical committee of a study where individuals are exposed to different levels of a compound if the compound is considered harmful, e.g. ionising radiation or smoking. On the other hand, people often, willingly or not, expose themselves to harmful factors and we thus have to rely on populations that have been exposed due to reasons beyond our control.

A specific feature of etiological epidemiological studies is that there is no possibility to tell if a specific carcinogen caused a cancer. Although molecular geneticists are trying to identify alterations that imply that a specific etiological agent caused a malignancy, however, no reliable tool is available today. We thus still have to depend on statistical differences between exposed and non-exposed populations. Models extrapolating risks for intermediate and high doses to low dose situations are necessary because of the inability of epidemiological studies to evaluate small effects of the exposure variables.

During the past decades extensive research on the long-term effects of ionising radiation has been conducted and epidemiological studies have contributed to our present knowledge. There are a number of populations under study and the most important source is the Japanese atomic bomb survivors. The data derived form this cohort has been used for determining exposure standards to protect the public and the workforce from harmful effects of ionising radiation.

The major exposure to low dose and low dose-rate radiation derives from medical tests, occupational, and environmental situations. The established model for determining carcinogenic effects at low doses in radiation protection is based on the hypothesis that the cancer incidence increases with radiation dose. Most national and international bodies have adopted a so-called linear no-threshold model. The major implication of the no-threshold model for stochastic effects is that all doses, regardless of how low they are, must be considered potential carcinogenic.

Important examples from major studies, strengths and weaknesses, will be presented and discussed.

The latest news from radiobiology – implications for risk assessment

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A change in radiobiological paradigm

A basic paradigm in radiobiology is that, after exposure to ionising radiation, the deposition of energy in the cell nucleus and the resulting damage to DNA, the primary target, are responsible for the harmful biological effects of radiation. The radiation-induced changes are thought to be fixed already in the first cell division following the radiation exposure and health effects are considered to result as a consequence of clonal proliferation of cells carrying mutations in specific genes.

These basic assumptions have recently been challenged by new research findings on radiation-induced genomic instability and bystander effects.

Genomic instability

It has been observed that the progeny of irradiated cells show occurrence of new mutations and/or new chromosomal aberrations or other genomic damage for many generations. Affected progeny also demonstrate high levels of lethal mutation, which may be measured as delayed reproductive cell death and/or delayed apoptosis. These delayed effects occur in cells that were not exposed to radiation.

Genomic instability occurs in the progeny of irradiated cells at a frequency that is several orders of magnitude higher than would be expected for a mutation of a specific gene. Therefore a mutation in a repair gene is not a likely explanation for the induction of genomic instability.

Induction of genomic instability has been demonstrated after very low doses of radiation. Using a microbeam exposure system, it has been shown that a single alpha particle is able to induce chromosomal instability in the progeny of cultured human cells. This is indeed the lowest possible dose of densely ionising alpha particles to cells.

Genomic instability is induced both by high-LET and low-LET radiation, but not all cell lines show this effect. Animal studies indicate that some mouse strains are genetically more susceptible to genomic instability induction than others. These strains also show a higher susceptibility to radiation-induced malignancy.

Existing data of radiation-induced genomic instability suggests that perpetuation involves non-traditional inheritance (stable non-mutational changes or epigenetic mechanisms). Increased oxidative stress seems to be a long term characteristic of the progeny of irradiated cells. The increased oxy-radical generation is maintained by a signal produced by the irradiated cells in the culture medium.

Bystander effect

Nontargeted or bystander effects are harmful changes in cells that were not directly hit by radiation but were nearby. The signal can be transferred via the culture medium ('clastogenic factors') or cell-to-cell communication (inhibition of cell communication prevents bystander effects).

Bystander effects are not new. Starting from the 1960's, there is extensive literature on clastogenic factors and other 'compounds' that stimulate or modify responses in cells that were not damaged. More recently, it has been shown that the changes in neighbouring cells include changes in gene expression and mutation induction. Moreover, it has been shown that the medium from irradiated human epithelial cells reduces the clonogenic survival of nonirradiated cells. This observation implicates a secreted factor that can kill undamaged cells and that appears to involve reactive oxygen species.

Modern microbeam exposure systems capable of exposing single cells or even defined cellular organelles to charged particles or ultra soft X-rays have also facilitated research on bystander effects. Irradiation of cellular cytoplasm with either a single or an exact number of alpha particles has been shown to result in gene mutation in the nucleus.

It has been proposed that the bystander response is the initiating event in radiation-induced genomic instability.

Implications for risk assessment

The main source of information on radiation-induced human cancer risk comes from epidemiological data on exposed populations. However, direct information is available only at relatively large doses, and mostly from low-LET radiations (X- and gamma rays). A linear extrapolation from this data is applied at lower doses, which are more relevant in terms of exposure to the general population and radiation workers. Additional extrapolation is applied to other radiation types. The shape of the dose response curve for cancer at low doses is a matter of constant debate. Arguments range from a threshold or even beneficial effect of small radiation doses (hormesis) to non-threshold supralinear responses (implying that small doses are more hazardous than previously assumed).

Radiation-induced genomic instability and bystander effects both indicate that deleterious effects can be observed in cells that were not irradiated. These effects are similar to those occurring in irradiated cells. Genomic instability and bystander effects are observed already after very low doses. In fact, some dose response data indicate that the relative contribution of these indirect effects as compared to damage caused by direct hits may well be more pronounced in the low dose region.

Indirect radiation effects such as genomic instability and bystander effects may be important early steps in the development of radiation-induced cancer. Understanding of underlying mechanisms may have profound consequences on the cancer risk assessment, and these events could potentially be incorporated into biological modelling of tumorigenic responses in the future.

Acknowledgements

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SESSION 2. PROTECTION OF THE ENVIRONMENT, INFORMATION AND ETHICS

Protection of the Environment: Current ICRP Work and EC-Funded Research

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Introduction

The requirement for assessments of the environmental effects of radiation, i.e. effects on non-human biota, is increasing due to growing public concern for environmental protection issues and integration of environmental impact assessments into the regulatory process. Thus, there is a strong need to establish a framework for the *assessment* of environmental impact of ionising radiation, as well as a system for *protection* of the environment from ionising radiation. These ambitions are reflected in a number of international efforts and various 'systems' have been proposed or are under development [1-7]. This paper considers the current discussions on environmental protection within the International Commission on Radiological Protection (ICRP), as part of the Commission's ongoing revision of its recommendations as laid out in Publication 60 [8]. Furthermore, the paper reviews work within the EC-funded FASSET (Framework for ASSessment of Environmental impacT) project [9]. The concepts developed both by ICRP and FASSET are similar, and the FASSET approach and results may illustrate how forthcoming ICRP recommendations could be turned into practical application.

The ICRP Task Group on Protection of the Environment

Previously, the ICRP has not explicitly dealt with protection of the environment, except in those situations where radionuclide levels in non-human organisms were of relevance for the protection of man [8]. Hence, there is little ICRP guidance as to how radiological protection of the environment directly should be carried out, or why. There are several reasons why ICRP now has considered it necessary to revise its position and future role with regard to protection of the environment, including:

- the need to demonstrate that radiological protection principles are consistent with existing international conventions and recognise the inter-dependence of man and other components of the environment;
- the necessity for operators and regulators to demonstrate compliance with existing international and national environmental requirements;
- the need to provide advice with respect to intervention situations; and
- the necessity to demonstrate explicitly how knowledge of the potential extent of effects of ionising radiation on the environment can be used to inform stakeholders.

To this effect, the Commission set up a Task Group in the year 2000 with the aim of developing a protection policy framework for environmental protection. The conceptual framework of this area of work would then feed into the Commission's recommendations for the beginning of the 21st century.

The Task Group has limited its scope to *effects of radiation on non-human biota*; effects on abiotic components of the environment have been excluded. Although it may be completely legitimate and justified for various reasons to consider also the abiotic components, it is highly unlikely that there will be *radiation effects* in those components under ambient radiation levels; hence, the limitation of scope.

A common approach for protection of man and the environment

For biotic components of the environment, it is knowledge of the dose to, and effects on, individuals that forms the initial basis for drawing conclusions on what actions that need to be taken in different exposure situations. For human beings, the Reference Man [10] is the primary reference for dose assessments, supported by a secondary set of data for a foetus, a child etc. Such data enable dose estimates to be made for 'hypothetical' or representative individuals under different circumstances of exposure. For environmental protection, a similar set of primary *reference fauna and flora*, or *reference organisms*, has been proposed as representatives of the biotic component of the environment [5], and this approach is also used in FASSET.

The selection criteria for reference organisms will include many scientific considerations, and will depend on to what extent they are considered to be *typical* representative fauna or flora of particular ecosystems. A reference organism does not represent an average or a sentinel organism, but would serve as a point of reference for making comparisons with other sets of information on other organisms. The ICRP Task Group intends to propose a stylised system for radiological protection of the environment, harmonised with the principles for the radiological protection of man along the lines described above. This system will be designed so that it can be integrated with methods that are already in use in some countries. The objectives of a common approach to protect man and the environment might be to safeguard human health by

- preventing the occurrence of deterministic effects;
- limiting stochastic effects in individuals and minimising them in populations;
- and to safeguard the environment by
- preventing or reducing the frequency of effects likely to cause early mortality, reduced reproductive success, or scorable DNA damage in individual organisms to a level where they would have a negligible impact on
- conservation of species, maintenance of biodiversity, or the health and status of natural habitats or communities.

A common approach to the achievement of these objectives could be centred on a set of reference dose models, reference dose per unit intake and external exposure values, plus reference data sets of doses and effects for both man and the environment. Such models have already been used and are being further developed with FASSET.

FASSET

FASSET is an EC 5th Framework Programme project, comprising 15 partners in Finland, France, Germany, Norway, Spain, Sweden and UK. FASSET aims at providing a formal framework for the assessment of radiation effects on biota and ecosystems, to assist decisionmakers and stakeholders when judging the environmental impact of radioactive contaminants. The project started in the year 2000 and will end in 2003. The project has the following practical objectives, also largely shared with a similar project oriented to the Arctic environment, EPIC (Environmental Protection from Ionising Contaminants in the Arctic):

- to provide a set of reference organisms relevant to different exposure situations.
- to provide a set of exposure models for the reference organisms;
- to examine data on biological effects on individual, population and ecosystem levels; and
- to review existing frameworks for environmental assessment.

3.1 Reference organisms

FASSET's working definition of the reference organism is: "*a series of entities that provide a basis for the estimation of radiation dose rate to a range of organisms which are typical, or representative, of a contaminated environment.*

These estimates, in turn, would provide a basis for assessing the likelihood and degree of radiation effects". The project has initially defined a number of candidate reference organisms on the basis of characterization of ecological characteristics, important pathways and radionuclide transfer processes in major European ecosystems (semi-natural ecosystems including pastures, agricultural ecosystems, wetlands, forests, fresh-water ecosystems, marine ecosystems and brackish ecosystems (Table 1).

Table 1. Candidate reference organisms identified from an exposure pathways analysis.

Terrestrial ecosystems			Aquatic ecosystems	
Soil	**Herbaceous layer**	**Canopy**	**Sediment**	**Water column**
Soil micro-organisms Soil invertebrates, 'worms' Plants and fungi Burrowing mammals	Bryophytes Grasses, herbs and crops Shrubs Above ground in-vertebrates Herbivorous mammals Carnivorous mammals Reptiles Vertebrate eggs Amphibians Birds	Trees Invertebrates	Benthic bacteria Benthic invertebrates, 'worm' Molluscs Crustaceans Vascular plants Amphibians Fish Fish eggs Wading birds Sea mammals	Phytoplankton Zooplankton Macroalgae Fish Sea mammals

The approach taken towards their selection should ensure that suitable reference organisms are available for a range of scenarios (chronic and acute exposure) and for different European ecosystems. The complete list of candidate reference organisms and the reasoning behind their selection is found in project Deliverable 1 [11], available on www.fasset.org.

Exposure analysis for various reference organisms

A number of radionuclide transfer models developed for different ecosystems will be used for tabulation of external and internal radionuclide concentrations. Furthermore, tabulations will be made to allow conversion of concentrations in the environmental media, and internally, to absorbed dose (rates); this will include consideration of the relative biological effectiveness (RBE) of different radiation types in the development of appropriate radiation weighting factors (w_r) for the organisms, endpoints and dose rates of concern.

The estimation of external exposure has initially focused on organisms in the terrestrial environment, and Monte-Carlo calculations have been used to estimate external exposure for various reference organisms. As an example, Figure 1 presents the dose conversion factor for a mole, which is exposed to a planar g-source on top of the soil. The dose conversion factor (DCF) increases in proportion to the g-energy, and decreases with increasing depth of the target due to the increasing shielding effect of the overlying soil layer. The differences in shielding are more pronounced for low energies.

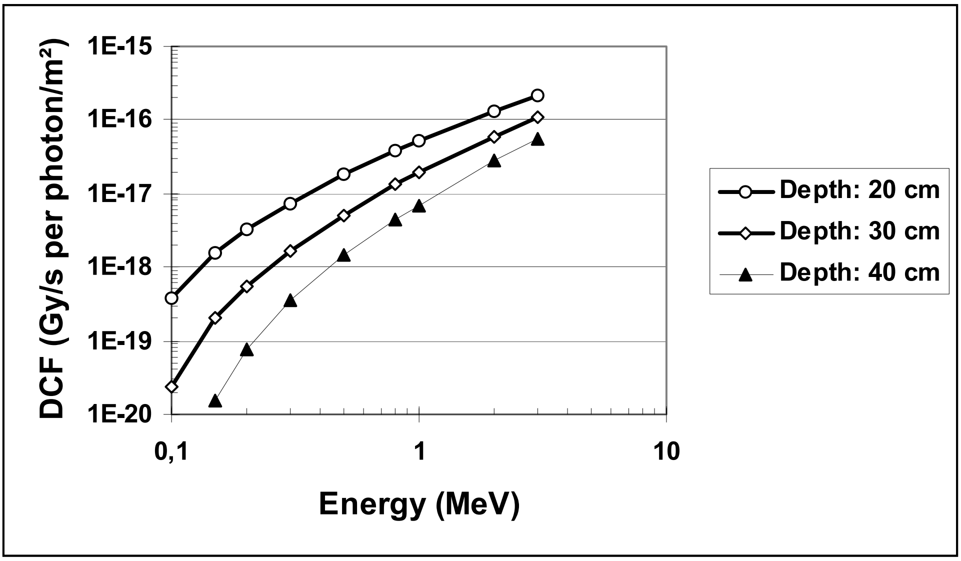

Figure 1. Comparison of the DCF for a mole as function of the g-energy and the depth of the target for a homogeneous planar source at the top of the soil.

For estimating internal exposures to biota, a set of organisms, sizes and energies have been defined that allow the assessment of exposures to a wide range of possible species. The most important quantity to assess internal exposures is the fraction of energy absorbed in the organism; this depends on the radiation type, the energy and the size and geometry of the reference organism. As a first step, a homogeneous distribution in the reference organisms will be considered. Additionally, radionuclide accumulations in specific organs, e.g., the thyroid and the gonads, will be simulated. Then, the dose for this specified organ will be calculated.

For the reference plants defined as herbaceous vegetation, shrub and tree, the exposure will be calculated for the meristem and the buds. These organs are characterized by very intensive cell division, which may cause high radiosensitivity. For the distribution of the radionuclides in the canopy, a distinction is made between a-, b-, and g-radiation due to their different ranges. For g-radiation the whole canopy is considered to be a homogeneously contaminated source of radiation. For high energy b-radiation, the irradiation of the target is also assumed to occur from a homogeneously contaminated canopy. However, due to the much shorter range of a- and low energy b-radiation, the irradiation from the external or internal contamination of the target organ has to be considered explicitly.

Biological effects of ionizing radiation

FASSET concentrates on four ‘umbrella effects’ that when expressed in the individual may not only affect the fitness of the individiual *per se*, but that may also be of significance at the population level:

- morbidity (including growth rate, effects on the immune system, and the behavioural consequences of damage to the central nervous system from radiation exposure in the developing embryo);
- mortality (including stochastic effect of somatic mutation and deterministic effects in particular tissues or organs that would change the age-dependent death rate);
- reduced reproductive success (including fertility and fecundity);
- cytogenetic effects (i.e. indicator of mutation induction in germ and somatic cells).

These four categories are not mutually exclusive, e.g., effects leading to changes in morbidity may result in a change in the age-dependent death rate, and an increase in mutation rate may lead to changes in reproductive success. They simply provide a convenient means of summarizing the available information in a structured way that is meaningful within the objectives of FASSET. In order to organize the available information on biological effects in a useful way for the framework, a database is being built; it is aimed at relating dose or dose rate to effects for the specific purpose of FASSET [12]. The collection of data for the database is currently ongoing.

Framework

The general structure of existing frameworks for environmental risk assessment has been considered to be appropriate also for FASSET; i.e., a division of the assessment into three stages: problem formulation, risk assessment, and risk management [13]. The risk management stage lies outside the scope of the project. The emphasis is on the development of tools and data for the risk assessment phase of this ecological risk assessment and management process. However, the construction of the framework must be flexible in order to take into account the various risk management options, as well as societal concern defined in the formulation stage, as these influence (and ultimately must make use of) the way in which a risk assessment is carried out. The way in which problem formulation is carried out in the different assessment systems studied differ, depending mainly on the different aims and philosophies of the assessments. The framework must be appropriate for problems of varying formulation, e.g., FASSET must:

- be able to take into account ongoing, past and future releases;
- be able to take into account chronic and acute effects;
- be appropriate for assessments carried out for various purposes, e.g., licensing, demonstration of compliance, assessment of accidents, and decisions concerning remediation.

Some of the elements of other frameworks will be included and appropriately adapted within FASSET, together with a justification for the approach taken. This information will be presented in Deliverable 2 of the project, due by the end of 2002, and will be publicly available at the project website.

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Risk perception and public participation processes

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Key words: risk communication, risk assessments, public hearings, transparency, nuclear waste repositories

Abstract

The paper summarizes the results of two studies and aim at presenting new developments in the area of risk communication research. The first study was conducted within a Nordic Nuclear Safety Research (NKS/SOS-1) project in 2000 where nuclear power personnel as well as politicians and administrators of a Swedish nuclear community participated in focus group discussions and questionnaire studies. The project aimed at pinpointing areas of perceived difficulty regarding the understanding and communication of nuclear safety issues. Central themes of the discussions included e.g. presentation of PSA-results, mutual expectations, roles, criteria for selecting and prioritizing abundant information, use of information channels, use of persons' knowledge and experiences, and use of terminology. Three main areas were outlined in the subsequent questionnaires, i.e. problems in information transmission, handling specific situations, and ways to improve the communication situation. Comparisons could be made between nuclear experts and politicians/ administrators on the basis of the resulting indices of the questionnaire, and the similarities of and differences between the groups are discussed in the paper. The study may have an impact on the local safety council of elected representatives in terms of agenda setting and internal communication procedures. The second study was ordered by the Swedish Nuclear Inspectorate (SKI) and asked for an evaluation of a number of public hearings conducted in six Swedish municipalities in 2001. The municipalities had previously been involved in feasibility studies for investigating their qualities for a possible future repository for high level nuclear waste. The hearings were planned and carried through in accordance with the ideas of a transparent communication and decision making process previously outlined within the RISCOM and RISCOM II-projects in Sweden and EU (see e.g. Andersson,

Espejo & Wene, 1998. Building channels for transparent risk assessment. SKI Report 98:6.Stockholm:SKI). Presentations by the implementer, The Swedish Nuclear Fuel and Waste Management Co (SKB), were followed by group discussions, and questions to SKB as well as to the nuclear authorities. Questionnaires asking for evaluations of the procedures showed satisfaction among all groups involved and pointed to a number of lessons learned. It was concluded e.g. that transparency in decision making can be enhanced by public hearings although a transparent process in not sufficient for public acceptance of a repository.

Existing foodstuff regulations promote risk miscommunication

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Abstract

The respect for, not to say the fear of, radioactive materials, in lay community is ubiquitous. The lay concern in a low-dose accident scenario, such as the Chernobyl accident fallout in the Nordic countries, makes radiation risk information in clear and understandable terms more important than intervention. The reason is that otherwise the negative non-radiological effects may easily outweigh the radiation carcinogenic and genetic effects. Evidently the radiation expert must in the low-dose accident scenario abandon ALARA and widen the focus to include also all conceivable non-radiological consequences, immediate or delayed, positive or negative.

In this contribution only a minor part related to this huge and complex risk optimising area is treated, namely food stuff regulations. It is concluded that the easy-to-control trading limits expressed in Bq/kg must be secondary to a radiation limit expressed in intake of activity per year (or a similar long period of time). Only the latter limit is risk significant enough to be launched and advocated by an authority responsible for radiation protection.

The Chernobyl accident came as a shock to all groups of people including media. The surprise combined with the unfamiliar technical terminology led to a unique mass-media feature reporting the accident, at least in the dominating news medium, the television. Compared to conventional accidents, much less of the news material was filtered through journalists, instead the radiation expertise were given plenty of live time on television to describe and comment the Chernobyl consequences. It is essential to have this feature in mind. It indicates that a pursuit of self-criticism is more relevant than trying to blame the mass media coverage, when summarizing the less than successful Chernobyl risk information. Blaming public ignorance is no good either, as any risk information must be on the level of the receiver. Radiation experts and authorities were not able to ease the public's anxiety, instead regulations and advice launched, added to it, and worse, messages and behaviour seemed confusing and contradictory. It is also important to note

that non-expert (in radiation that is) authorities, literate and well-educated people were confused. A comic strip by Stroyer in the leading morning newspaper Dagens Nyheter illustrates the ambiguity. In this single strip, published less than a week after the accident, a character with a gas-mask says “No danger”. Another grade comes from the Swedish Gallup Institute interviewing a representative group of Swedes on two occasions in May 1986. On May 1, a few days after the accident, 22 % of Swedes suspected that authorities were hiding the dangers of radioactivity. On May 25, despite the frequent appearance of radiation experts on prime time television, the corresponding figure was 35 %.

This loss of trust is serious. In any emergency situation in which people are, or believe they are, in danger, it is of prime importance from an information point of view, to build up a trustful relation. People take advice only from informers that they trust and believe have relevant knowledge. For most other information they close their ears. Considering the fact that radiation experts and authorities in 1986 presented their views honestly and with the best of intentions, indicates that there must have been something wrong with their message. A major cause for the loss of credibility lies with the foodstuff regulation, which rightly caused a lot of indignation among people. Instead of emphasising that the yearly-consumed activity is the primary concern, the intervention level expressed in Bq/kg was the frontier quantity, misleading the public to believe that this level was the borderline between safety and harm. And worse, no distinction was made between basic food and e.g. spices, the same intervention level of 300 Bq/kg was enforced. Today we have differentiated trading limits and they are essential and necessary in order to create confidence when buying and selling food. But differentiated trading limits do not resolve the problem at stake, the credibility between the radiation expert and the public. In the absence of a risk relevant radiation limit expressed as an annual limit of activity, any trading limit will in the public eye be the difference between safe and danger.

In order to minimise future lay/expert distrust concerning radioactivity, it is essential that the focus be on yearly intake of activity instead of trading limits. All listings of trading limits, including Codex Alimentarius, must firmly state limits (or recommendations) of yearly intakes of activity and stress that these limits are primary. Besides being risk relevant, primary food levels in Bq/y open up for people’s own decisions concerning consumption. It is of great psychological value that the individual himself can decide what to eat, especially for farmers, hunters etc living partly on non-commercial food. The risk information quality from the public point of view, will improve significantly if it is made clear that the primary limits enforced contain a margin of safety and give a dose comparable to a typical medical X-ray investigation.

Optimering av strålskyddet efter ett Tjernobylliknande nedfall

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Under våren 2002 har det i Sverige diskuterats om de svenska gränsvärdet för Cs-137 som SSI beslöt om våren 1986 (300 Bq av Cs-137 per kg) var för lågt satta. I debatten får man intrycket att gränsvärdet var mycket för lågt men den vetenskapliga bevisföringen för att så är fallet är minst sagt undermålig.

Optimering av strålskyddet är en viktig strålskyddsåtgärd och skall egentligen ske även om gränsvärdet medger högre halter. Jag skall försöka beskriva hur jag skulle vilja se en optimering av de åtgärder som bör vidtagas i händelse av en ny deponering av radiocesium. Detta gör jag med en erfarenhet från hur Tjernobylnedfallet har påverkat bl. a. jordbruk och renskötsel och hur frågan har hanterats av olika myndigheter i Sverige från 1986 och framåt. Jag anser att jag dessutom kan göra detta helt oberoende av olika myndigheter som var inblandade i hanteringen av Tjernobylnedfallet – det är inte något försvarstal för myndigheter. Mitt resonemang baseras på dagens kunskap om riskerna med inducering av cancer orsakat av joniserande strålning d.v.s. ICRPs riskbedömningar från 1990 som måste anses vara den bästa riskuppskattningen som vi har med dagens kunskap. I första hand kommer jag att diskutera kostnader för några motåtgärder som kan vara aktuella och komma fram till vilka gränsvärden som kan anses vara vetenskapligt och ekonomiskt motiverade. Eftersom mjölken är den absolut viktigaste transportören av radiocesium till människan diskuterar jag radiocesium i mjölk.

Det första man måste bestämma sig för är vilken risknivå som kan anses vara acceptabel när det gäller kontaminering av vår föda. I livsmedelsfallet anser man att för mutagena och cancerogena kemikalier är en risknivå på 1 på 100.000 till 1 på miljonen acceptabel. När det gäller joniserande strålning – i detta specifika fall strålning orsakat av intag av radionuklider från Tjernobyl - anges ofta 1 på 20.000 som en acceptabel risknivå. Ett ofta förekommande påstående är att man accepterar en större risk när det gäller kemikalier i vår föda jämfört med strålning från radioaktiva ämnen – är det någon skillnad är det snarare så att vi tolererar en större risk när det gäller strålning. Trots allt utgår jag från ambitionen att hålla risknivån på 1 på 20.000 som varande en acceptabel risk. Det motsvarar i stort sett risken att under ett år omkomma i en trafikolycka. Skillnaden är att Tjernobylnedfallet var en påtvingad risk medan trafikolyckor åtminstone delvis innebär ett frivilligt riskåtagande. En effektiv stråldos av 1 mSv ger just risken 1 på 20.000 – alltså skulle 1 mSv per år kunna anses vara en acceptabel extra stråldos. Det finns dock många olika åsikter om detta. Det motsvarar ett intag av Cs-137 på omkring 75.000 Bq under ett år. Om vi skulle få ett nytt Tjernobylliknanade nedfall har vi dessutom Cs-134 med och detta innebär att intag av omkring 40.000 Bq av Cs-137 under det första året skulle ge 1 mSv. Vi kan alltså acceptera intag av 40.000 Bq av Cs-137 under första året efter ett Tjernobylnedfall. Jag förenklar resonemanget genom att inte ta med jodisotoper i diskussionen. Nästa steg är att fastställa ett gränsvärde för Cs-137 i mjölk alternativt att optimera insatserna för att reducera överföring av Cs-137 till människa via födan. Detta bör ske genom att balansera kostnaden för en viss motåtgärd med den nytta, d.v.s. den stråldos man kan ta bort genom att sätta in en viss motåtgärd. Ur beräkningssynpunkt är det enklast att gå ut från att vi helt enkelt häller bort mjölken. Det är som regel den dyraste åtgärden man kan vidtaga – det finns billigare alternativ t.ex. installning av mjölkkorna med utfodring med icke kontaminerat foder. Om vi nu utgår ifrån en riskbild enligt ovan och sätter ett pris av 10 miljoner per människoliv skulle det vara relevant att betala 500 kronor för varje mSv man kan ta bort med lämpliga motåtgärder. För att få 500 kronor måste mjölkproducenten leverera 167 liter mjölk eftersom producentpriset på mjölk är omkring 3 kronor. Alltså, 167 liter mjölk får innehålla 40.000 Bq av Cs-137 d.v.s. 240 Bq per liter mjölk. Observera att vi inte tagit hänsyn till att mjölken kan innehålla I-131 och vi endast har diskuterat den dyraste motåtgärden – gränsvärdet borde egentligen vara något lägre.

Min slutsats är att om man utnyttjar det bästa kunskap vi har just nu kommer man fram till att gränsvärdet för Cs-137 skulle vara 240 Bq per liter mjölk. Det man kan diskutera i mitt resonemang är vad som kan anses vara en acceptabel risk. Min åsikt är att vi inte kan acceptera en större risk än 1 på 20.000 om vi vill att strålskyddet skall arbeta med ungefär likvärdiga risker som används i kemikaliefallet. Det är alltså en myt som tyvärr är mycket spridd att strålskyddet nu skulle ligga i första ledet när det gäller att undanröja risker i vårt samhälle. Strålskyddet har historiskt sett legat i frontlinjen men detta tycks inte gälla numera åtminstone när vi diskuterar vår föda.

Ett problem som uppkommer något senare efter ett nedfall av Cs-137 är hur man skall göra med gränsvärdet för kött. Priset på kött är betydligt högre än priset för mjölk. Beräkningarna blir något mer komplicerade därför att man kan sätta in kraftfulla motåtgärder som reducerar Cs-137 halterna i köttet. När det gäller slaktdjur finns alltid möjligheten att sätta in lämpliga motåtgärder före slakt och det är kostnaden för dessa som skall jämföras med den reduktion av stråldos man kan uppnå. Det är troligt att man kommer fram till att gränsvärdet borde vara högre än vad vi kan acceptera för mjölk. B. Åhman har dock visat att balansen mellan kostnad och nytta var relativt god under 1986/87. Om ingen motåtgärd skulle ha gjorts när det gäller renar hade den potentiella kollektivdosen orsakad av renköttets innehåll av Cs-137 i Sverige uppgått till 193 manSv. Den verkliga dosen blev 3 manSv d.v.s. man tog bort 190 manSv genom att kassera ett stort antal renkroppar. Det överraskande är att om man jämför kostnaden för alla åtgärder - ca 95 miljoner kronor med nyttan d.v.s. antal sparade manSv gånger 500.000 kronor per manSv får man 85 miljoner - alltså. en överraskande god balans. Statistiskt sett motsvarar borttagande av 190 manSvmellan 9 och 10 fall av dödlig cancer och för att förhindra detta kanske det är värt att satsa 100 miljoner kronor. Personligen anser jag att SSI inte behöver gå ut och be om ursäkt för att gränsvärdet hamnade fel. Om det skulle ske en ny Tjernobylhändelse skulle jag dock gärna se olika gränsvärden för mjölk och kött men på en betydligt lägre nivå än vad EU föreslagit. Om man utnyttjar det kunnande som trots allt har byggts upp efter Tjernobyl skulle det finnas stora möjligheter att göra mera selektiva ingrepp - och därmed billigare - än vad som gjordes 1986.

Considerations on ethics and decision making in radiation protection

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Radiation protection is not only science

The system of radiation protection is under discussion due to the plans of the ICRP to change their general recommendations from 1990. To replace the present generally used system, the new system should be widely acceptable in terms of underlying ethical views and the level of protection provided.

Radiation protection aims to prevent harmful effects from exposure to radiation. To be useful and accepted as a basis for risk management the radiation protection system should address risk assessment methods and reasoning and justification of the desired level of protection. Risk assessment is based on science. Assessments are made through dose calculations and the use of dose-effect relationship. However, the magnitude of risk is only one factor in decision making. Other aspects which affect the acceptability of risks and decision making on protective measures include such values and targets as equity, individual autonomy, protection of future generations and environment, practicability of actions and proper use of available resources.

Individual and collective doses describe different aspects of the risk

Providing protection to each individual is generally seen as an ethically correct way of action. Individual dose limits serve this target. At the same time, ethical views of many people require that the total harm caused by the activity taken should be small and the balance of risks and benefits should be pursued. Accordingly, they think that both individual and societal aspects of the risk matter should be taken into account when forming risk management policy. This view implies also that risks are only acceptable if reduced to as low as reasonably achievable (ALARA).

It is desirable to have all relevant characteristics of the risk available for decision making. Therefore, it is important to look at the activity and the risk caused by it from various perspectives. The use of the linear no-threshold dose-effect relationship (LNT) is widely accepted as a risk assessment tool in radiation protection. This method is often thought to provide cautious assumptions for risk evaluations. Most people think that using LNT is in line with precautionary principle and therefore it is ethically correct. Accepting LNT implies that the overall risk of an activity is described through the collective dose.

Collective dose is a proper tool for evaluation of the ALARA target. Using collective dose estimates allows different aspects of the risk management to be taken into account:

The public. The public is exposed to many different sources. Radioactive releases from a source can cause in distant places and times exposures which are in most cases small. Due to the stochastic nature of the risk from small doses the number of exposed people is a crucial factor in evaluating the harm. It may be that for a large number of exposed people the overall risk is not considered to be negligible although individual doses might be small. Therefore collective dose is a useful tool when looking for optimum solutions for controlling radiation sources.

Workers. The collective dose suits particularly well to measure the effectiveness of ALARA-programmes designed to reduce the exposure of the workers. It serves as an indicator of the quality of radiation protection.

Medical. It is obvious that collective dose is useful for the evaluation of mass screening programmes. Medical use of radiation causes a significant part of the exposure to population. Again, collective dose can be used to evaluate the situation and its´ changes.

Decision making in complex situations

There are many varying views on the acceptable level of risk and the aspects to be taken into account in development of risk management policies. Therefore, clear reasoning concerning these aspects should be given in the radiation protection system recommended for the bases of radiation protection management.

Decision making in radiation protection should be based on information on the harm caused in order to address the actual problem clearly. To understand the relevant importance of risks, comparisons with other risks, e.g. background radiation and its variations, are useful. However, only comparison with background doses with no description of the associated harm distorts the focus of the decision making. Also, additional doses caused by human activities are different from background radiation exposure because activities are planned and therefore the doses can be controlled.

Uncertainties add to the complexity of the situation to be evaluated and make decisions more difficult. For example, information on long-term effects of an activity and resulting collective doses from it is more and more uncertain the longer the relevant periods to be covered are. Uncertainties should be described and included in the factors addressed but this does not mean that uncertain information should be ignored.

Targets for risk management change with time. On a broad scale, risks of people have been reduced when comparing the present and past. Changes in risk levels and other developments of society may have an influence on the perception of acceptable risks in the future. We should also consider the possibility that risks that are tolerated today will not be tolerated in the future. Also, risks that persist for long periods because of contamination of large areas, or even worldwide, may limit activities in the future due to combined additional risks. The radiation protection system does not aim to limit the long term contamination of the environment if radioactive releases of long-lived radionuclides that cause very small individual doses are not counted at all.

The ethical goal of equity can be seen to include several aspects that need to be recognised when considering risks from a radiation source. One factor is obviously the dose received by an individual or a group of individuals. An example of another factor is the share of benefits and burdens. In many cases, those who are affected through radiation exposure or through economic consequences caused by a radiation source do not receive any benefit from it. A further aspect affecting the equity of individuals is that some people have a possibility to influence the decision making while others have not. In many cases those who have no benefit from the actions causing radiation exposure to them or cannot influence the decision making are future generations or live in distant areas. This reasoning reminds us of the complexity of situations to be managed and suggests that it is not fair to reduce the equal treatment of individuals to a requirement of equal radiation exposure limit from a source. Consequently, ethical requirement to safeguard each individual's rights implies particularly careful reasoning for protection of those who cannot take part in the process of decision making.

Acceptable for whom?

One of the three main requirements of the present radiation protection system is optimisation. In relation to exposures from any particular source protection should be optimised in order to keep the magnitude of individual doses, the number of people exposed and the likelihood of incurring doses all as low as reasonably achievable. The economic and social factors should be taken into account, within the restriction that the doses to individuals delivered by the source be subject to dose constraints. Requirement to optimise doses is a wise principle in radiation protection. It provides flexibility in decision making while promoting careful thinking in implementation of radiation protection due to the requirement of finding out what is reasonable. Optimisation urges also for improvements in practice as it implies that development of methods and changes in availability of resources should be taken into account.

Problems may arise when implementing the optimisation because it is not always clear which levels of exposures are reasonable and for whom exposures should be reasonable. Evaluations of risks include identification of relevant issues and definition of acceptable levels of risks. For acceptable management of risks this process should be a matter of all interested parties, particularly those affected by subsequent decisions. However, it is impossible to get consent of each individual who has an interest in the matter. Involvement of representatives of stakeholders is the way to incorporate various views in the process in practice.

Different and even conflicting opinions are presented on radiation protection. Democratic processes are designed to come to a solution in such a case. Open participation, equal availability of understandable and full information to all those taking part in the process is a prerequisite for fairness of the process.

Posters

2a Radioprotection from environment to man - ENVIRHOM

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Presented at the XIIIth ordinary meeting of NSFS, Turku, Finland 25.-29.2002, by Tua Rahola, STUK/Laboratory of Radiation Hygiene and member of the ENVIRHOM scientific international evaluation committee.

ENVIRHOM is a research programme towards the improvement of risk assessment for situations of chronic exposure to radionuclides. The aim is to study the bioaccumulation of radionuclides in situations of chronic exposure of ecosystems and members of the public.

The project was initiated by IPSN, now IRSN, in France. The programme is designed for a duration of 10 years. International collaboration is planned and a proposal for an integrated project (BIOME) will be submitted to the EC within the 6th FP.

Scientific basis of the system of radioprotection

The system of radioprotection is based on knowledge of the effects of ionising radiation on population groups exposed to relatively high doses and dose rates (Hiroshima and Nagasaki). The effects corresponding to lower levels, characteristic of the exposure of the public and most nuclear industry workers are extrapolated on the basis of a linear no threshold dose/response model.

Research carried out in the field of radioprotection for low doses have focused mainly on the exposure of workers in the nuclear fuel industry. The concepts and the models available for this category of the population are judged adequate for the degree of accuracy required in that the rules applied are very conservative. However, these models correspond to particular situations of exposure (acute respiratory contamination) that are not at all representative of those encountered by members of the public (chronic exposure by ingestion). The public are above all exposed to discharges from nuclear fuel cycle plants during normal operation and those from waste storage sites. The models applicable for members of the public should therefore be distinct from those of nuclear industry workers and take into account several specific parameters:

- **contamination levels**
- **contamination pathways**; (workers: acute inhalation, members of the public chronic ingestion)
- **the physical and chemical forms of the radionuclide and the complexity of the source term;** (workers : known physical and chemical form for the radionuclide, members of the public: physical and chemical forms occurring in the different physical compartments of the environment vary widely and transfers of between and within physical and biological compartments of ecosystems modify these forms; this context is totally ignored in the dosimetric impact assessment models for the general public
- **the multipollution context;** human activities generate the dispersal of a range of pollutants as trace elements in the environment; the possible synergy or inhibition of the effects of radionuclides in conjunction with other pollutants are totally ignored
- **the physiological condition of the individual;** this may vary widely from one person to another and may be very different from that of the ICRP's "standard man".

The project is organised in four themes, characterisation of biokinetics and bioaccumulation after chronic exposure, biological effects (excluding cancers), dosimetric consequences for both the environment and man and the mechanisms of bioaccumulation as illustrated in Figure 1. The radionuclides listed to be studied are uranium, three transuranic elements, ^{129}I, ^{210}Po, ^{232}Th, ^{99}Tc, ^{135}Cs and ^{137}Cs. The biological models will be plants (wheat), invertebrates (mussels, crustacea), freshwater fish (salmonoid), marine fish (sea bass and plaice), rodents (mouse and rat) and -conditionally- an aquatic angiosperm (Potamogeton) and prey fish (carp) and also other organisms. The multipollutant environment will be investigated in experiments involving heavy metals (Cd and Zn), organic pollutants (polychlorinated biphenyl, PCB), euthrophication (carbonenriched sediments, fertilisers) and acidification (lowering the pH in an extreme case to a value of 3).

Validation is planned in "cage experiments" within natural ecosystems. The acquired knowledge will also be used for a systematic effort to improve the background to radiological protection of humans, to improve the methodology for environmental impact assessments, and to improve comparative assessment of environmental risks associated with radioactive and non-radioactive environmental contaminants with emphasis on chronic low dose rate exposure.

The project was started in 2001 and the programme is still to be more closely defined taking into account also the suggestions for improvement by the evaluators. Starting with the extensive expertise and resources of IRSN it will after some time be necessary to involve outside collaborators. There is growing activity in the field of radiation protection of the environment. There is an ICRP Task Group on Environmental Protection and an EU 5th Framework Programme project "Framework for Assessment of Environmental Impact" (FASSET). The FASSET project is presented at this meeting.

ENVIRHOM is the first and very extensive project including also man as part of the environment. The results obtained will be of great value to the ICRP for the revision of the recommendations planned to be finalised by 2005.

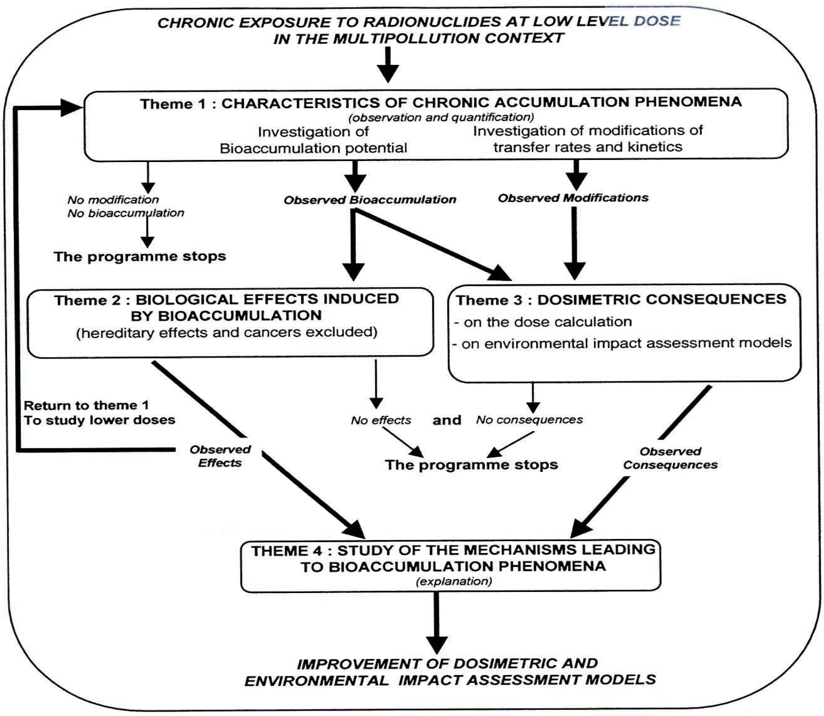

Figure 1. Interlinking the four research themes constituting the backbone of the proposed research programme.

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2b On *de minimis*, collective dose and the new ICRP recommendations

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Introduction

The main news in ICRP's proposal to a new radiation protection philosophy is the change of focus from minimising the collective dose, to a focus on individual protection. This abstract examines some of the argumentation that has been used in the debate.

The individual focus of protection in the new proposal is reflected by the maximum additional dose that any individual would be exposed to. ICRP suggests that effective doses above 100 times the average global natural background level cannot be justified in any normal situation. Doses below 1/100 of the average global natural background level would on the other hand be totally excluded from the system of protection. As it is still assumed that even small doses pose a risk this is clearly an introduction of a *de minimis* dose, though ICRP do not use the term explicitly. The introduction of a *de minimis* dose is discussed below.

The new proposal also includes a requirement to optimise protection for a given source. This is necessary to achieve *the best level of protection that is reasonable,* which seems to be a concept similar to the ALARA concept. Doses below the *de minimis* level are excluded from the optimisation. However, the optimisation is made for each (potentially) exposed individual and on selected groups, and not on the total collective dose. The new proposal gives a very limited role to the collective dose concept. The arguments for this are discussed below.

On a *de minimis* dose

By declaring a risk to be de minimis *it is usually implied that the risk is so small that it should be ignored. That is, a* de minimis *risk is so small that it is beyond concern, or equivalent (from a decision perspective) to no risk at all. (Peterson 2002)*

ICRP still presume the probability of harm to be proportional to the dose, and without a threshold. From this follows that exposure to even a small dose implies exposure to a risk. If this dose (risk) is small enough, it is regarded to be *de minimis*. ICRP sets the level of the *de minimis* dose (and corresponding risk) to about 1/100 of the global average of the natural background radiation. This would mean a *de minimis* annual effective dose of about 0.01 mSv.

The argument ICRP give for choosing this level is that the proposed dose is trivially small compared to the natural background; "It is not possible to receive a dose of 0.01 mSv in isolation, and, therefore, the associated risk is for increments above background and to associate a risk with 0.01 mSv is meaningless." (Holm 2002)

ICRP also seem to motivate the level of the *de minimis* risk by public opinion; "This level of imposed or involuntary risk is about the most that has been judged as being tolerated by members of the public." (Clarke 1999)

A general criticism to *de minimis* risks is that though people may find some risks negligible, it is not certain that they are willing to subject themselves to other risks, no matter how small, especially if they have nothing to gain from it. It is also hard to see how naturally occurring risk levels are relevant for establishing a *de minimis* risk level. We may have a high, naturally occurring, risk level, which causes large detriments, but would we not choose a lower risk level, and corresponding detriment, if we had the choice? Another objection is that comparisons of natural with anthropogenic risks are inappropriate as we cannot do anything about the natural risks, but that alone is not a reason for introducing additional risks.

Even after establishing an individual *de minimis* risk level, the question remains if the corresponding societal risk is also *de minimis*? The proposition that this is the case is fundamental in the new ICRP recommendations: "If the risk of harm to the health of the most exposed individual is trivial, then the total risk is trivial – irrespective of how many people are exposed." (Clarke 1999)

A problem with this argument is that a risk, which is trivial for everybody, need not necessarily be trivial to the society. This is articulated by Lindell (2000); "... the fact that each individual is satisfied with his own protection does not necessarily mean that he or she is satisfied with the situation; they may dislike the idea that the odds are that, after all, someone might, unnecessarily be harmed."

There seems to be something missing in the argumentation for a societal *de minimis* risk. If we could reduce or avoid a societal risk very cheaply and easily, would we not do it, even if the individual risk were *de minimis*? It is hard to see how practices like unnecessary radioactive food additives could ever be *de minimis.* What is missing is the discussion of justification. A notion of this also seems to be held by ICRP; "Additional doses from *justified practices* far below the natural annual dose should not be a matter of concern to the individual or the society." (Holm 2002, my italics)

This suggests that a *de minimis* dose should not be discussed without its context.

On collective dose

The new proposal from ICRP gives less importance to the collective dose concept. This has made commentators point out that the collective dose, as a measure of the expected detriment, is a direct consequence of the linear, no-threshold assumption. However, there can still be good reasons not to use collective dose as a measurement of the total detriment. Some of these are discussed below.

The main argument for not using collective dose seems to be the claim that collective dose calculation gives exaggerated estimations of the detriment. The argument that the collective dose is exaggerated merely because the integration is carried out including very small doses to a very large number of people is either yet a variation of the *de minimis* argument, which is discussed above, or a non-expressed disbelief in the linear, no-threshold assumption.

An egalitarian objection is that the collective dose conceals information about the distribution of individual doses in both magnitude and time; "a large dose to a few people is not equivalent to a small dose to many people". It is true that collective dose should not be used exclusively as a measure of the detriment, as this could result in unjust consequences for extremely exposed people, but this does not prove that the collective dose cannot be a relevant, albeit insufficient, measure of the total detriment.

That the use of collective doses encourages that too much money is being spent to achieve low dose levels is not a good argument as optimisation, by definition, does not go beyond the point where the detriment can be reduced with "reasonable cost". Dunster (2000) argues that the collective dose also encourages the use of automatic, rather than well considered, procedures for taking protection decisions. Whether or not this is true is an empirical question.

Another argument is that global contamination or geological timescales give absurd values of the collective detriment. But the mere fact that the exposure of many people also makes the detriment large is not a good argument to discard the collective dose. It is true that calculations of the collective detriment should be used with care because of the uncertainties involved, especially when integrating over large time spans, but this is already recognised in the existing recommendations.

Large inherent uncertainties in the calculations of collective detriment do not necessarily render the collective dose useless in all applications. Instead these uncertainties could be expressed and discussed explicitly. Consideration should also be taken to the precautionary principle, which calls upon us to calculate with unlikely or uncertain scenarios.

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SESSION 3A

New Euratom research programme 2002-2006

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Introduction

The European Union funded research is carried out under two distinct programmes: the European Community Framework Programme (EC Programme) covering research, technological development and demonstration activities; and the Euratom Framework Programme covering research and training activities in the nuclear sector. At the moment, the fifth framework programmes are going on.

The fifth Euratom Research and Training Framework Programme (FP5) is approaching to its end. FP5 formally started in 1998 and will cease at the end of 2002, although several projects started during these years will continue in a couple of the coming years. FP5 covers research, training and technological development in the areas of thermonuclear fusion, nuclear fission and radiological sciences. For many years, nuclear fusion research has been fully integrated at the European level and it has been carried out as a co-operation with Japan, the Russian Federation and Canada. There are no open calls for proposals in the field of nuclear fusion.

The sixth Framework Programmes (EC and Euratom) [1, 2, 3] for 2002 – 2006 will start in the beginning of 2003. EU's funding for the EC Programme during the next four years will be 16 270 M •, which is more than ten times bigger than that of the Euratom Programme (1 230 M•). Thematic priorities and structure of the EC Programme is shown in Table 1. This paper addresses to the sixth Euratom Programme (FP6) and to the instruments used in implementation of the FP6. The both framework programmes use the same instruments and are tools of the new concept of the European Research Area (ERA).

The 6th Euratom Framework Programme (FP6)

Content and structure of the new Euratom Programme is shown in Table 2. Controlled thermonuclear fusion and direct funding of the Joint Research Centre take the major part of the funding of EU. The rest of FP6 can be called as a Fission Programme, which has only two thematic priorities: *Management of radioactive waste* and *Radiation protection*. Other parts of the Fission Programme deal with *New innovative ways of producing nuclear energy, Education and training,* and *Safety of existing nuclear installation*.

Table 1. Content and structure of the 6th Framework Programme of the European Community for 2002 – 2006.

Focusing and integrating Community research:	**13 345 M€**
• **Thematic priorities:**	11 285
- Life sciences, genomics and biotechnology for health	
- Information society technology	
- Nanotechnologies and nanosciences	
- Aeronautics and space	
- Food quality and safety	
- Sustainable development, global change and ecosystems	
- Citizens and governance in a knowledge-based society	
• **Specific activities covering a wider field of research:**	
- Policy support and anticipating scientific and technological needs	
- Horizontal research activities involving SMEs	
- Specific measures in support of international co-operation	
- Non-nuclear activities of the Joint Research Centre	
Structuring the European Research Area:	**2 605 M€**
• **Research and innovation**	
• **Human resources**	
• **Research infrastructure**	
• **Science and society**	
Strengthening for foundation of the European Research Area:	**320 M€**
• **Support for co-ordination of activities**	
• **Support for the coherent development of policies**	
TOTAL	**16 270 M€**

Geological disposal of spent fuel and other long-lived wastes on one hand and *Partitioning and transmutation* on the other hand will be the main topics in the waste management research. The high priority of waste management research is understandable because final decisions on how to handle and dispose of spent nuclear fuel and other long-lived wastes are getting more and more topical in the Member States. Public information and communication in connection of waste disposal will also play an important in future research activities.

Table 2. Content and structure of the 6th Framework Programme of Euratom for 2002 – 2006.

Priority thematic areas of research:	**890 M€**
• **Controlled thermonuclear fusion**	750
• **Management of radioactive waste**	90
• **Radiation protection**	50
Other activities in the field of nuclear technologies and safety:	**50 M€**
• **Innovative ways of producing nuclear energy**	
• **Education and training**	
• **Safety of existing installations**	
Nuclear activities of the Joint Research Centre	**290 M€**
TOTAL	**1 230 M€**

Quantification of risks at low and protracted doses will be the keynote in radiation protection research. This topic is of special importance since a lot of basic knowledge on radiation effects at low doses at the cellular level is still missing. Molecular biology and epidemiological studies will play the major role in this issue. *Protection of the environment* is another important topic in radiation protection research and it is presumable that it will have a high priority in the forthcoming work programme. Other topics which are expected to be included in radiation protection research are *Emergency management,* exposure to *Medical use of radiation and natural sources of radiation,* and *Occupational exposure to radiation*.

What is new in the 6th Framework Programme?

The content and especially implementation of FP6 will deviate considerably from the previous FPs. The main focus of FP6 is the creation of a *European Research Area* (ERA) [4] as a vision for the future of research in Europe. At their summit in Lisbon in March 2000, heads of state and governments called for better leveraging of European research efforts through the creation of ERA. The FP6 is the financial instrument that will help make the ERA as a reality.

In practice, implementation of ERA means *simplified management procedures* of research, *reduced number of projects* and hence *bigger projects*, *increased flexibility and management autonomy* within the projects, *changes in Commission's monitoring procedures* (from detailed monitoring of inputs to more strategic monitoring of outputs), and *new instruments to implement FP6*.

Integrated projects (IP) [5] and *Networks of excellence* (NoE) [6] will be the most important instruments in implementation of FP6. IP's are designed to generate the knowledge required to implement the priority themes by integrating the critical mass of activities and resources to meet the clearly defined scientific and technological objectives. In the Fission Programme this means that the number of projects will be reduced from the present about 200 to some 10-20 in FP6. NoE's are designed to network (integrate) existing and emerging research activities to provide European leadership and to be a world force in a particular topic. The basic concept for NoE is thus to bring together research actors, which are already funded at a national or regional level.

Also traditional instruments will be maintained. *Stairway of excellence* is designed to smooth the transition from the "old" to the "new" instruments (present concerted actions) and is especially directed to smaller research actors (including SME's) and to participants from the candidate countries. *Specific support actions* are planned to complement FP6 and to prepare future R&D activities (means conferences, meetings, studies, awards, etc.).

Participation rights will also be extended. The candidate countries associated the FP6 can participate the Programme under same conditions as EU Member States. Also European research organisations (e.g. CERN, ESA, EMBO etc.) will be treated as any entity from the Member States. Changes in partnerships within established IP's and NoE's can quite freely be decided by the partners.

Time table for implementation of Euratom FP6

Approval of FP6 (both the EC and the Euratom Programmes) was given 3 June 2002 by a co-decision of the Council and the European Parliament. Council Decision adopting the Euratom FP6 was given 7 August 2002.

On 20 March 2002 the Commission launched an Invitation to submit Expression of Interest (EoI) [7] as "an opportunity for Europe's research community to help prepare for the first calls of FP6". To the deadline 7 June 2002 about 200 EoI's have been received for the two thematic priority areas of the Fission Programme (Management of radioactive waste and Radiation protection), and about 50 EoI's for other activities. The EoI's were assessed in July and the results of assessment will be published in September 2002. All received EoI's will also be published on the CORDIS web-site. Purpose of the EoI's was to gain information on areas where the research community is prepared to make proposals for IP's and NoE's. The Commission will use the results of the assessment in the preparation of the Work Programme and the first call for proposals, which is expected to be published by the end of 2002.

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[5] Provisions for implementing Integrated Projects, http://europa.eu.int/comm/research/fp6/networks-ip.html.
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SESSION 3. NATURAL RADIATION

Natural Radiation - Exposures and Controls

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Exposures to natural radiation world wide and in the Nordic countries:

Table I. UNSCEAR 2000. Average radiation dose from natural sources.

Source	Worldwide average (mSv/y)	Typical range (mSv/y)
External exposure		
- Cosmic	0,4	0,3-1,0
- Terrestrial gamma	0,5	0,3-0,6
Internal exposure		
- Inhalation (radon)	1,2	0,2-10
- Ingestion	0,3	0,2-0,8
Total	2,4	1-10

Table II. Nordic Flagbook 2000. Average radiation dose from natural sources, mSv/y.

Source	Finland	Sweden	Denmark	Norway	Iceland
External exposure					
- Cosmic	0,3	0,3	0,3	0,3	0,3
- Terrestrial gamma	0,5	0,5	0,3	0,5	0,2
Internal exposure					
- Inhalation (radon)	2,0	1,9	1,0	1,7	0,2
- Ingestion	0,3	0,3	0,3	0,4	0,3
Total	3,1	3,0	1,9	2,9	1,0

Occupational exposures world wide to natural radiation sources:

Table III. UNSCEAR 2000. Occupational radiation exposure.

Source/practice	Workers (thousands)	Average dose (mSv/y)
Man-made sources		
- Nuclear	800	1,8
- Industrial	700	0,5
- Defence	420	0,2
- Medical	2320	0,3
- Education	360	0,1
Total, man-made	4600	0,6
Enhance natural		
- Air crew	250	3,0
- Mining (- coal)	760	2,7
- Coal mining	3910	0,7
- Mineral processing	300	1,0
- Above ground (radon)	1250	4,8
Total, natural	6500	1,8

ICRP recommendations 1959-1990 regarding control of exposures from natural radiation sources:

IRCP Publication No. 2 (1959).

- Natural background radiation varies considerably from locality to locality and the doses it contributes to the various organs are not well known.
- If maximum permissible limits recommended by the Commission included background radiation, the allowable contribution from man-made sources – which are the only ones that can be controlled – would be uncertain and would have to be different for different localities.

ICRP Publication No. 26 (1977)

- …the Commission's recommended dose limits have not been regarded as applying to, or including, the "normal" levels of natural radiation…
- …man's activities can increase the "normal" exposure to natural radiation…

o Mining
o Flight at high altitudes
o Building materials
o Houses

ICRP Publication No. 60 (1990)
- Occupational exposure: include exposures to natural sources only in the following cases:
o Workplaces identified by regulatory agencies
o Operation and storage of materials with significant traces of natural radionuclides
o Operation of jet aircraft
o Space flight
- Public exposure: +/- practices, use of intervention

- Radon in dwellings: Action levels

Requirements to Members States of the European Community in the Basic Standards Directive 1996 (EU-BSS):

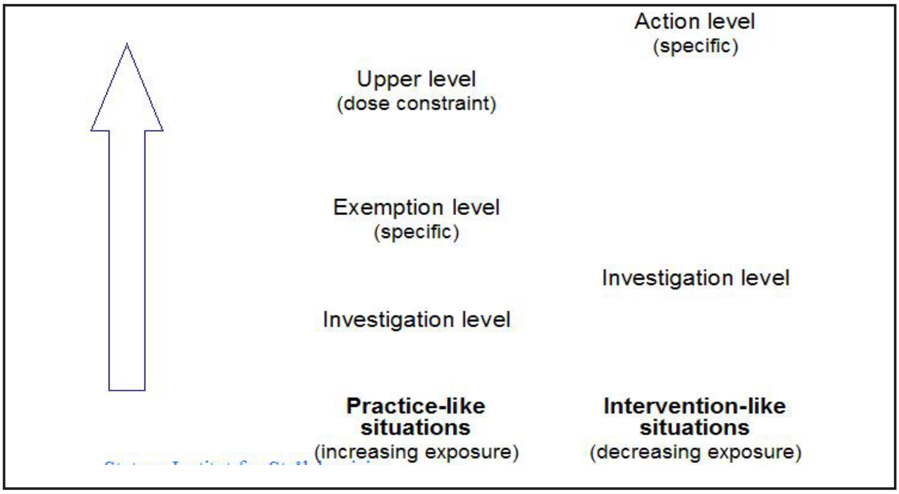
EU-BSS (1996)
- Do not apply to:
o Radon in dwellings (Recommendation, 1990)
o Natural levels of radiation
o Radionuclides in the human body
o Cosmic radiation at ground level
o Radiation from the undisturbed earth's crust
- Applies to:
o Practices and intervention
o Work Activities with Natural Radiation Sources (Title VII)

EU-BSS Requirements for Work Activities

- Work Activities declared to be of concern
o Monitoring of exposure
o Implementation of all or part of the requirements for practices
o Implementation of all or part of the requirements for intervention

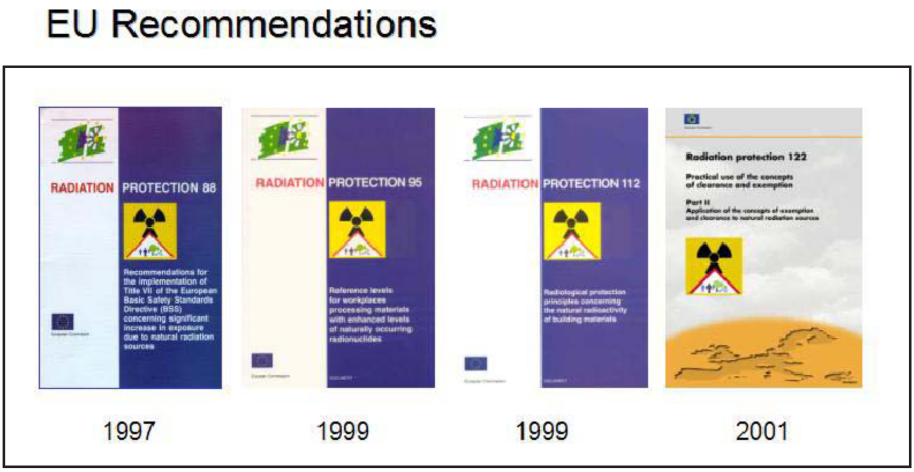
Nordic recommendations regarding control of exposures to natural radiation sources:

Flagbook – Hierarchy of recommended levels



EU recommendations regarding control of exposures to natural radiation sources:

- Radon in dwellings (Official Journal, 1990)
- Radon in workplaces (RP 88, 1977)
- Exposure of air crew (RP 88, 1997)
- Workplaces – Materials and Residues (RP 88, 95, 1997, 1999)
- Building Materials (RP 112, 1999)
- Exemptions and Clearance (RP 122 part II, 2001)
- Radon in water supplies (Official Journal, 2001)



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Radon-safe building in Finland in 2002

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Abstract
Slab-on-ground is the prevalent substructure in Finnish low-rise residential buildings. The gap between the foundation wall and floor slab is the main entry route of radon-bearing air from soil into living spaces. Performing the sealing work and installation of radon piping are the key measures of radon-safe building in houses with slab-on-ground. Sealing measures have been carried out in a minor part of houses due to the difficulties in the sealing practice presented in the current guidance. A new construction for an airtight joint between the foundation wall and floor slab has been studied in several test houses. Based on the results, a new guidance for a wide use in Finland is under preparation. In the new sealing practice bitumen felt will be installed underneath the floor slab in direct contact with concrete slab. The municipal authorities require increasingly radon-safe construction.

Introduction
High radon concentrations of indoor air in low-rise residential buildings form an important national health problem in Finland. Radon-bearing soil air is the main reason to elevated indoor radon concentrations of dwellings in Finland. According to the resolution of the Ministry for Social Affairs and Health in 1992, the indoor radon concentration should not exceed 400 Bq/m^3 in existing houses, the target for new construction being less than 200 Bq/m^3. The average indoor radon concentration in single family houses is close to 150 Bq/m^3, hence the target concentration of 200 Bq/m^3 is very demanding. In wide areas of the Southern Finland 200 Bq/m^3 is exceeded in more than 30 % of houses.

Decrease in the use of crawl space in house foundations has increased the indoor radon concentration of Finnish housing stock during last decades. The prevailing type of foundation is slab-on-ground (Fig 1). In this type of foundation the flow of radon-bearing air from soil into living spaces through gaps between the foundation wall and floor slab should be prevented through special measures. During the last ten years radon-resistant new construction has become more and more common in areas with high indoor radon concentrations and also all over the country. However, the practices have been variable and defective. Radiation and Nuclear Safety Authority (STUK) has performed questionnaire studies in order to explore the prevention techniques and municipal practices (Ravea and Arvela 1997, Voutilainen et al. 1998, 2000). This study presents results of these questionnaire studies and of a new development project "Radon-safe foundation, moisture prevention and air exchange in a healthy building" (RAFO), funded by Finnish Technology Agency. The aim of the project was to develop simple construction practices which can be utilised for prevention of these leakage flows and which form simultaneously quality moisture prevention for the foundation structures (Arvela et al. 2002). Several building companies participated in the project. The project aimed also at development of radon technical practices of the companies.

Materials and methods

The questionnaire studies aimed at exploring how the builders had followed the guidance given on radon-safe building (Ministry of Environment 1994). This guidance has been studied and developed also in the RAFO-project. The guidance aims at radon-safe construction of slab-on-ground, cellar and hill-side foundations. The guidance can be applied to more than 80 % of new single family houses because of the prevalence of these substructure types. The radon-safe construction comprises two measures. First air-tightness should be achieved using bitumen felt and elastic sealants. Second preparatory perforated piping should be installed.

The data regarding the preventive measures taken in 300 dwellings was ob-tained from a questionnaire study carried out in 1995-1996 (Ravea and Arvela 1997). The subject group was 300 house-owners having taken preventive measures in order to achieve a low indoor radon concentration. The second questionnaire study (Voutilainen et al. 1998, 2000) aimed at exploring the radon prevention practices in municipalities with high indoor radon concentrations. The questioning was made by phone to 95 municipal building authorities and to 55 health authorities in municipalities.

In the RAFO-project the construction of foundations of 20 houses was followed up. All measures related to the air-tightness of the foundation as well as difficulties and drawbacks were documented. When the houses were ready and in normal use indoor radon concentration, air exchange rate and depressure were measured. The effect of activated sub-slab radon piping on the drying process of the floor slab was tested in two houses.

Results and discussion

Questionnaire studies

The questionnaire study addressed house-owners showed that the most common measure was installation of the sub-slab piping, with no sealing work. However, although the target level of 200 Bq/m^3 was exceeded, no fans had been normally installed at the time of the study. The target level of 200 Bq/m^3 was still exceeded in 40 % of the houses.

Sub-slab piping with an operating fan provides an efficient preventive measure. In 80 % of houses with sub-slab piping conne-ted to an operating fan, radon concentration was below the action level of 200 Bq/m^3. In houses with piping but no fan, the corresponding fraction was only 45 %. The corresponding median values of radon concentration in these houses were 55 and 220 Bq/m^3, respectively. Sub-slab piping without a fan had no remarkable effect on radon concentration. In houses with crawlspace and edge-thickened slab, radon concentrations were low. The choice of foundation system thus significantly affects the indoor radon concentration.

Sealing was performed in a low number of houses although it is an essential part of radon-safe building. Sealing measures were carried out in about 30 % of the houses and in many of these houses the sealing was not carried out along the quideline. However, through performing careful sealing work, good results were achieved without the acti-vation of the sub-slab piping.

The questionnaire study addressed to municipal authorities revealed that only in 30 % of the municipalities studied the authorities demanded or recommended to react to radon in the whole area of the municipality. In 70 % of the cases they demanded radon-safe measures at least in a certain restricted area. The most important prevention measure for houses with slab-on-ground, radon tight foundation construction was recommended only in a few municipalities. Installation of the preparatory sub-slab piping was the normal measure taken by the builders.

RAFO-project

The radon-tight construction practices published in the guidance of the Ministry of Environment (1994) were studied in the project. In addition a new more easy-to-do construction was studied. The construction given in the guidance requires a high level carefulness. Along the results of this study the sealing of the joint between the foundation wall and floor slab fails often in real building site conditions. A new construction for a radon-tight joint between the foundation wall and floor slab was developed for houses with slab-on-ground or houses with basement. The results were very promising. In the new construction a wide (0.5-1.0 m) strip of bitumen felt will be installed above the foundation wall. The other edge

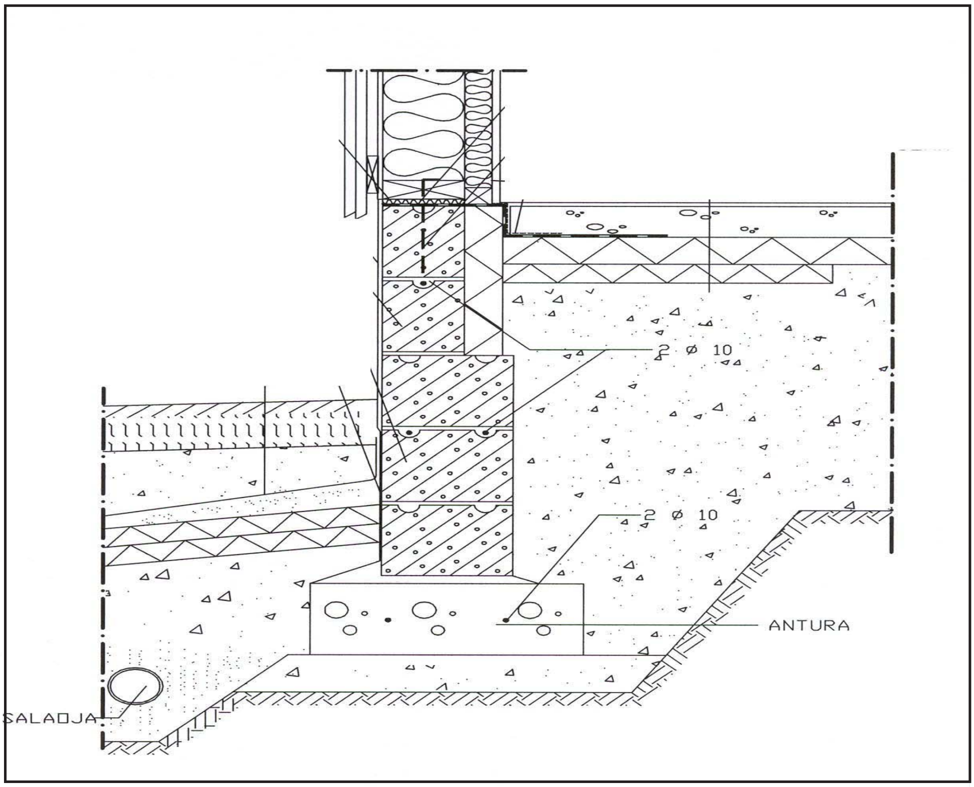

Figure 1. Installation of bitumen felt to the joint of foundation wall and floor slab.

of the strip will be installed underneath the concrete slab in direct contact with the casting. The key advantage is the exclusion of the need for elastic sealant. The tests made in laboratory ascertained that the adhesion of the felt to concrete was good.

Indoor radon concentrations in 7 test houses varied in the range of 20-60 Bq/m^3. Five of the seven houses located in an area were 50 % houses exceed 200 Bq/m^3. In the 8th house where the new method was tested radon concentration was 220 Bq/m^3. However, this was a low value in an area where 90 % of houses exceed 200 Bq/m^3 (median 590 Bq/m^3). Lack in installation of bitumen felt and sealing of lead-throughs was considered as the main reason to remaining radon leakage.

In the sealing of lead-through the methods presented in the guidance of the Ministry of Environment showed to be working. The practical experiences showed that special attention should be paid to the development of the guidance of sealing the water pipes and the electric wiring which are installed in protective piping beneath the floor slab. Above-slab installation would be preferable.

The results showed that airtight construction is needed in the whole building area. Omitting the sealing work for example in storerooms or in rooms for house technology may lead to increase in radon concentrations in living spaces. In these auxiliary spaces radon concentration was in some cases 5000-10000 Bq/m^3, when the sealing was omitted.

Summary

- The municipal authorities have required radon-safe construction often only in restricted areas with high indoor radon concentrations
- Since 2000 in many municipalities new local building regulations require radon-safe construction in the whole municipality.
- STUK has recommended radon-safe building practices in the whole country
- A new construction for an airtight joint between the foundation wall and floor slab was developed for houses with slab-on-ground and houses with basement. The construction prevents the flow of radon-bearing air from soil into the house. The results will be utilised when improving the present guidance material.
- The bitumen felt used in the construction works also as a qualified moisture insulator.
- A new guide to be published in the Finnish RT-Building Information File is under preparation.

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Mitigation of radon in drinking water of Finnish drilled wells - Present state and Cost-benefit analysis

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Introduction

The annual collective dose from natural radioactivity of ingested drinking water in Finland is estimated at 260 manSv. The users of drilled wells receive about half of the collective dose, although their proportion of the population is only about 5%. Radon causes approximately 70% of the ingestion dose. Apart from ingested water, consumers are exposed to radon and its decay products when radon is released into indoor air in situations like showering and washing cloths and dishes. The lung cancer risk from inhaled decay products of radon depends strongly on the smoking prevalence in the exposed population. To combine the both exposure paths, ingestion and breathing, it is more feasible to calculate the number of cancer deaths rather than doses. In addition to the number of annual cases, the cost per averted cancer case is influenced by the total costs of the chosen removal method, frequency distribution of radon concentration in the wells and the action level applied.

When the action level for radon in drinking water in a private well is exceeded, STUK recommends joining the municipal water system or a water co-operative as a primary alternative. These options assure both the quality of drinking water and a proper wastewater disposal. However, neither of these alternatives is commonly available in sparsely populated areas. Another viable choice is to dig a well into the soil. If none of these alternatives is possible, two methods can be used for removing radon from drinking water: aeration and activated carbon filtration. If the filtration method is used without adequate shielding, an additional dose may incur due to external gamma radiation emitted by the short-lived decay products of radon that are retained in the carbon. STUK recommends that an activated carbon filter unit not be installed inside a dwelling, but be placed in a separate building or in the subsurface service space beside the drilled well.

Until now, about one thousand drilled wells are equipped with radon removal units. This is about ten per cent of the total need for mitigation. This necessitates further use of effective radon removal methods or alternative water sources in reducing potential cancers.

Materials and methods

In Finland the distribution of radon concentration in drilled wells is nearly lognormal with an excess number of very high concentrations. According to our data, the mean and the median are 570 Bq/L and 140 Bq/L, respectively (1, 2). In order to calculate the number of wells and well water consumers with exceeding action levels, as well as the mean concentration of radon before and after mitigation, the data were weighted as described in previous papers (1,2). The number of permanent users of approximately 80,000 drilled wells is estimated at 300,000, the number of consumers per well being 3.6 on average. The number of consumers was assumed larger in this study than that in the previous papers.

The averted dose from ingested water was calculated using the dose conversion factor of 3.5×10^{-9} Sv/Bq (3). For the assessment of the number of annual cancer deaths, we applied the ICRP estimate of a 5% increment of fatal cancer incidences caused by 1 manSv. Concerning inhaled, waterborne radon, the dose approach does not take into account the strengthening effect of smoking on the lung cancers. Therefore, in STUK, a national estimate of 200 annual fatal lung cancer cases caused by the Finnish mean indoor radon concentration of 120 Bqm^{-3} has been made using the results of both epidemiological studies among miners and the nation-wide epidemiological study (4). We used this estimate and the transfer coefficient of 10^{-4} (the ratio of indoor radon concentration due to water use to the radon concentration in water) to calculate the cancer cases due to inhaled waterborne radon (3).

The effectiveness of radon mitigation depends on the type of the removal unit, the professional skill of the seller and the plumber, and the quality of the water. With proper equipment and installation, an efficiency of 98–99% can be achieved by using either aeration or activated carbon filtration methods. In practice, this is not always the case. Therefore lower values (95 and 90%) are considered in this paper, too.

The expenses of radon removal are comprised of acquisition and installation costs, and the annual costs of operation and water quality monitoring. We calculated repayment periods of 10, 20 and 30 years to obtain estimates for the total annual costs. Plausible service lives are 20 years for the aeration device and 30 years for the activated carbon filtration unit as well as for a new well dug into the soil. Total annual costs were calculated by dividing the investment costs by the service life and discounting the annual operation and monitoring costs during the repayment period into the present value using a discount rate of 3%. As to public water supply, the annual operation costs were calculated by using a reference value provided by the Finnish water and waste water works association (FIWA) and by discounting it into the present value. The connection fee was added to the total costs by assuming a real interest of 3 % (5).

The action level applied in Finland for radon concentration in water for private wells is 1,000 Bq/L. This is set as a quality recommendation in a decree of the Ministry of Social Affairs and Health that is based on the 2001/928/Euratom recommendation of the European Commission.

Results

Average acquisition and installation costs of an aeration device are about twice of that of an activated carbon unit. This difference will be partly compensated by the higher operating costs due to changing the carbon batch every other year (Table 1). The longer service life of the activated carbon filtration device,

Table 1. Typical costs of removal methods and alternative water sources (ranges in parenthesis).

Cost factor	Costs (euro)			
	Activated carbon filtration	Aeration	New well dug into the soil	Joining into public water supply
Installation (or digging the well, connection fee to join public water supply)	400 (200–600)	800 (500–1,000)	1,500 (1,000–2,000)	900 (0–17,000)
Acquisition of equipment	1,000 (500–2,000)	2,500 (1,500–4,000)	800 (500–1,000)	
Operation and water quality monitoring (per year)	160	130	60 (20–100)	190
Total costs per year, 10-year repayment period	300 (200–400)	400 (300–600)	200 (100–300)	
Total costs per year, 20-year repayment period	200 (150–250)	**300 (200–350)**	120 (70–200)	
Total costs per year, 30-year repayment period	**150 (130–200)**	200 (150–250)	**100 (50–150)**	**150 (70–600)**

Table 2. The effect of action level applied on the cancer cases averted.

Action level of radon Bq/L	Number of wells exceeding the action level	Number of Consumers using water exceeding the action level	Mean radon concentration after radon removal Bq/L	Number of cancer cases averted by the mitigation of 95% effectiveness
0	80,000	300,000	0	10
100	50,000	200,000	50	10
300	30,000	100,000	90	9
1000	10,000	40,000	200	7
3000	3,000	10,000	300	4

however, limits the annual cost of the filtration method to only half of that of the aeration method. The costs of a new well dug into the soil are even 50% lower than the costs of filtration. The repayment period of the founding expenses of a well was assumed as 30 years.

Annual costs from drinking water distributed by waterworks to a reference house with annual consumption of 180 m^3 of water are about 200 euro. When the fees on the 30-year period are discounted as previous, and 3 % of the connection fee is added, the mean value of annual costs is 150 euro.

With a fixed distribution of radon concentration in water, the number of cancer cases averted depends on the action level applied. When mitigating all wells exceeding the action level of 300 Bq/L most of the cancers caused by radon could be averted (Table 2). The effect of the mitigation effectiveness is not noteworthy, as can be seen in the Figure 1.

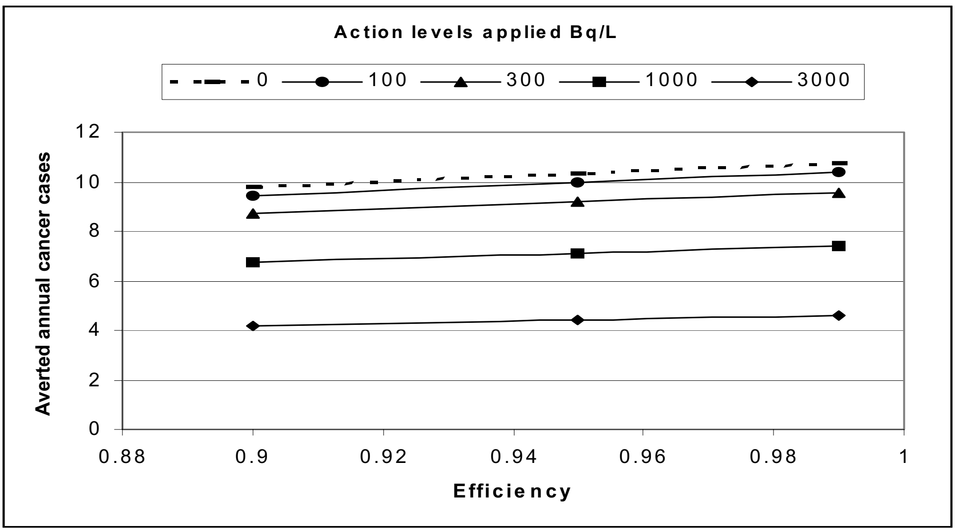

Figure 1. The dependence of the removal efficiency on the number of annual averted cancer cases when applying four different action levels

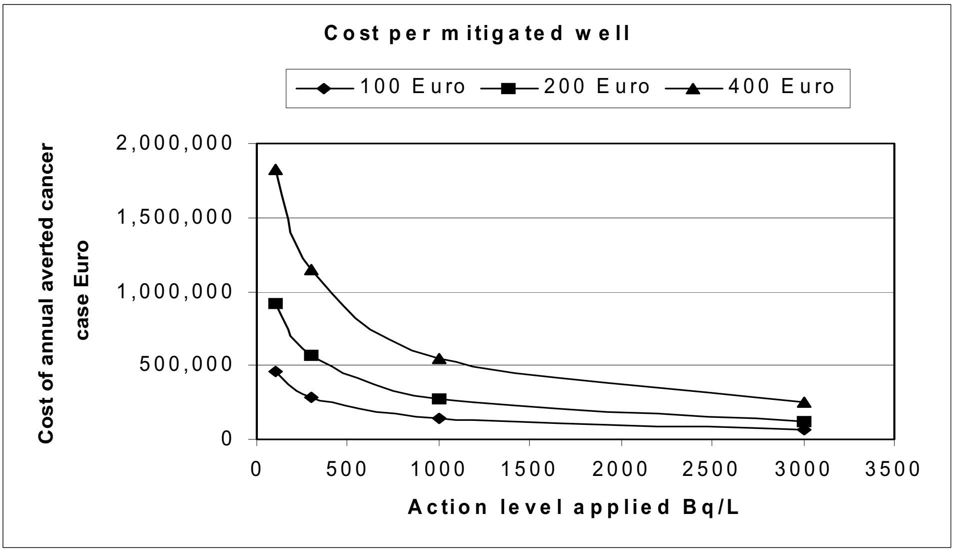

Figure 2. Costs of an averted cancer case as a function of applied action level when the annual mitigation costs vary from 100 to 400 euro.

When the action level of 1,000 Bq/L for radon is used, the costs of an averted cancer case are about 300,000 euro (100,000–600,000) (Figure 2). Decreasing the action level to 300 Bq/L would double the costs of a saved life, while increasing the number of the averted annual cancer cases from 7 to 9. On the other hand, the action level of 3000 Bq/L would halve the costs when the number of averted cancer cases would decrease from 9 to 4.

Discussion

The 1000 radon removal systems installed by summer 2002 will save 0.7 cancer cases annually. However, the number of averted cases and, furthermore, calculated costs per averted cancer case are sensitive to the risk model. If the risk model from inhaled radon introduced by BEIR 6 were used, the number of averted cancer cases would be more than twofold and thereby, the costs per averted cancer case would be less than half of those calculated in this study (6).

According to this study, when applying the current action level of 1,000 Bq/L the costs per an averted cancer case by mitigation of radon in water are rather high, typically 300,000 euro. Nevertheless, considering the number of lives saved and the costs, the present action level is well justified. The corresponding costs for an averted cancer case saved by mitigating indoor radon are about 100,000 and 20,000 euro for action levels of 200 and 800 Bq/m^3, respectively (7). The current action level for indoor radon is 400 Bq/m^3. The mitigation of indoor air in dwellings is thus more cost-effective than the mitigation of radon in drinking water.

However, the consumers in Finland are more motivated for mitigation of radon in drinking water than in indoor air. This may be due to the tradition of monitoring of drinking water rather than indoor air, and the available commercial radon removal methods for private wells. The methods for mitigation of indoor radon are more complicated and few competent companies are working in the field. Therefore, the authority should make more effort to provide people with more information about cancer risks of indoor radon. The companies should also find motivation to further develop radon mitigation products for indoor air and market them widely.

Digging a new well into the soil may be the most economic solution for the mitigation of radon in drinking water. However, there is a risk that the water from the dug well contains contaminants (e.g. iron and humus) that require appropriate removal units, too. The second cheapest options are activated carbon filtration and joining into the public water supply, and the most expensive is the aeration method. However, a consumer making a choice may not see the options in this order, and all of them are not necessarily available. In sparsely populated areas a public water supply may not be available at any price. A well dug into the soil requires suitable soil with adequate aquifers. The radon removal equipment requires special space and accessory, which may not be easily available and may cause extra costs depending on the house and the well. The activated carbon filtration system has to be installed outside the dwelling so that the residents are not exposed to external radiation. The aeration equipment has to be installed into a room with a floor drain, electric wall socket and heating.

Water quality and proper wastewater treatment are guaranteed best by joining the public water supply. Monitoring of private well water is often more or less at the responsibility of the resident. The convenience and high quality of water offered by public water supplies make them a competitive choice even when compared to temporarily cheaper options.

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Classification and treatment of LSA-sludge from the Brent Spar

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Abstract
Assessments of the levels of radioactivity in sludge from the Brent Spar storage tanks have been performed. Based on field and laboratory measurements, only 44 drums (of a total of 2268 drums) were identified as drums with ^{226}Ra-concentrations above the exemption level of 10 Bq/g given by the Authorities. Field measurements showed satisfactorily agreement with laboratory measurements. The drums containing radioactivity concentrations below the exemption level were classified as non-radioactive and shipped to Sløvåg Industriservice AS for incineration. Only small amounts of radioactivity were found in the ashes after incineration. The concentrations in the offgas and the outlet water were below or close to the detection limit for gamma spectrometry. Incineration of sludge with activity concentrations below the exemption level was found to be a well-suited method for disposal.

Introduction
On request from Wood-GMC AS in 1999, Institute for Energy Technology (IFE) assessed the levels of radioactivity in sludge from the oil storage tanks on Brent Spar. The sludge was contained in 2268 standard-size steel drums at Raunes Quay at Vats, Rogaland, Norway.

The assessment was carried out in two stages:

I) Survey of the distribution and levels of radioactivity throughout the storage facility, including estimation of the within-drum variation in activity concentration (homogeneity test) [1,2].

II) Final assessment including identification of the drums containing sludge with radioactivity concentrations above the exemption level given by the Norwegian Radiation Protection Authority [2,3].

IFE was also asked to carry out verification measurements during incineration of the sludge that had been classified as non-radioactive [2,4].

Legislation

A preliminary exemption level of 10 Bq/g for ^{226}Ra, ^{228}Ra or ^{210}Pb was given by the Norwegian Radiation Protection Authority (NRPA) in 1995 [5]. Waste from the petroleum industry with activity concentrations exceeding this level for one of the radionuclides mentioned shall be regarded as radioactive waste and handled as such. This exemption level is in accordance with international recommendations [6,7].

Stage I

Sampling and measurements

The sludge was contained in numbered standard-size steel drums stored in containers. Based on knowledge of the origin of the sludge, it was assumed that the activity concentration in drum number *x* would be similar to the activity concentration in drum number *x+1*. For the purpose of establishing regions of drum numbers with expected high, intermediate or low activity concentrations, one sample was collected in one drum in each storage container. All samples were taken below the "water table" inside the drums to avoid sampling of oxidised and semi-dry material. Four additional samples were collected in selected drums for test of the variation in activity concentration with the drums.

A total of 127 samples were collected from 107 of the drums. All samples were packed in plastic containers and analysed with high-resolution gamma spectrometry. Based on earlier measurements of the Brent Spar sludge, ^{210}Pb could be excluded from the assessment, as its presence was only as a daughter of ^{226}Ra. ^{226}Ra was analysed directly by its 186 keV gamma line and the results were corrected for self-absorption, which was 12% for the counting conditions in question. ^{228}Ra was analysed through its daughter ^{228}Ac.

Results and recommendations for Stage II

The results from Stage I revealed no drums with an activity concentration exceeding the exemption level of 10 Bq/g. The mean activity was found to be 5.9 ± 1.5 Bq/g and 1.6 ± 0.5 Bq/g for ^{226}Ra and ^{228}Ra respectively. The ^{226}Ra:^{228}Ra ratio was found to be 3.8 ± 0.5 with three outliers all high in ^{226}Ra. The mean within drum variation in activity concentration for ^{226}Ra was found to be 6.7% with the highest variation being 13.0%. It was therefore concluded that the homogeneity of the material was sufficient for conclusive determinations with respect to the exemption level based on one sample per drum for measured activity levels below 8 Bq/g. Additional sampling might be necessary for activity levels exceeding 8 Bq/g.

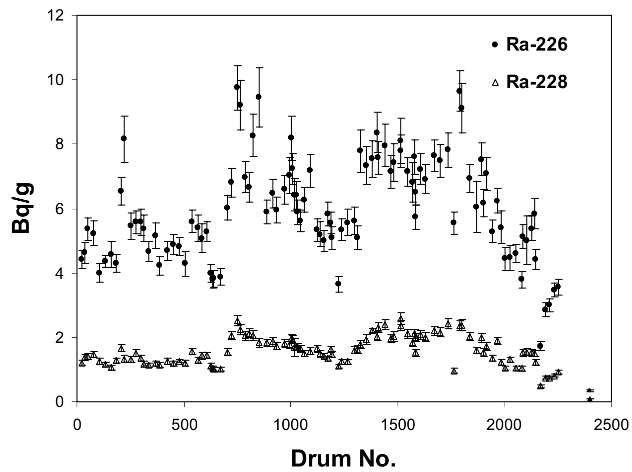

Figure 1. Specific activity of ^{226}Ra and ^{228}Ra vs. drum number. Results from Stage I.

Viewing the results as a function of drum number (Fig.1), there appeared to be three drum number intervals with activity concentrations that might exceed the exemption level. All drums in this region should therefore be sampled during Stage II. In all other regions it was not deemed necessary to sample and measure the activity concentration in each and every drum.

An action level of 8 Bq/g were proposed for the field measurements in Stage II. All samples exceeding this limit should be brought back to IFE for further measurements using high-resolution gamma spectrometry. Measurements above this action level from drums in expected low activity regions should lead to sampling of ten drums above and below.

Stage II

Sampling and measurements

The work in Stage II was performed according to recommendations given in Stage I. Samples were collected after the same procedure as in Stage I and packed in standard size plastic containers. Screening measurements of the sludge were carried out at site using a hand held monitor (NE Electra with a beta sensitive BP4-probe, Fig. 2). The measurements were made with a fixed geometry between the probe and the samples to ensure proper analytical quality. The instrumental set-up was calibrated using the results and the ratio between the two radium isotopes obtained in Stage I.

Each sample was measured for one minute and the background count rate automatically subtracted. The background count rate was measured

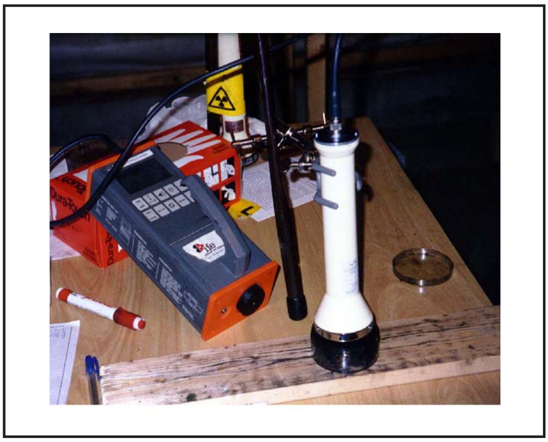

Figure 2. Experimental set-up at Vats.

regularly, and if it had changed significantly, a new background count rate was determined. All samples measured to have activity concentrations above 8 Bq/g were sent to IFE for possible re-measurement using high-resolution gamma spectrometry.

Assuming a reasonable correlation between the field measurements and the "true" values it was decided to analyse the samples brought back to IFE in batches with decreasing order of activity until no more samples within one batch were found to have activity concentrations exceeding the exemption level. The rest of the samples were then assumed to contain less than 10 Bq/g of ^{226}Ra. ^{226}Ra was analysed directly by its 186 keV gamma line and the results were corrected for self-absorption.

Results and conclusions

A total of 1266 measurements were made at Vats (the results are shown in Fig. 3). Based on the screening measurements samples from 203 drums were found to have a ^{226}Ra concentration above 8 Bq/g and consequently brought back to IFE. At IFE 164 samples, divided into six batches, were measured with high-resolution gamma spectrometry.

The criterion for selection of the drums with concentrations above the exemption level was defined to be: "Analytical value + total analytical uncertainty $\geq$ 10 Bq/g".

Based on this criterion, a total of 44 drums were identified and classified as "above exemption level"-drums. These drums were shipped back to the UK. The mean ^{226}Ra-activity concentration of all the sludge material was found to be 6.0 ± 0.6 Bq/g.

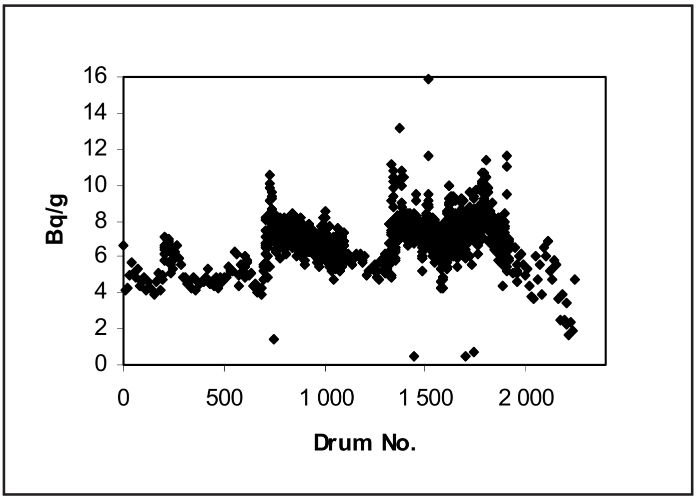

Figure 3. Specific activity of ^{226}Ra vs. drum number. Results based on fieldwork and measurements at Vats – Stage II

On average, each drum contained 230 kg of sludge (data from Wood-GMC AS). Total activity of ^{226}Ra was then estimated as:

- Total ^{226}Ra activity in all Brent Spar sludge: 3.1 ± 0.3 GBq in 2268 drums
- ^{226}Ra activity in "above exemption level"-drums: 0.10 ± 0.01 GBq in 44 drums
- ^{226}Ra activity in "below exemption level"-drums: 3.0 ± 0.3 GBq in 2224 drums

There was good agreement between the results obtained in Stage I and the results from the screening measurements in Stage II. All "above exemption level"-drums identified in Stage II were found in the high activity regions identified in Stage I. Also the correlation between the field measurements and the "true" values were fairly good. The average discrepancy was about 8 %, with a maximum of 27.9%. Considering the simplicity of the method used in field, this must be regarded as a satisfactory result.

Incineration process

The remaining 2224 drums were shipped to Sløvåg Industriservice AS for incineration. To obtain optimal conditions for combustion, the sludge was mixed with drill cuttings at a ratio of 1:9. An evaluation of the expected activity levels in exhaust gases, water and ashes had been made before the verification work began [8]. The radioactivity in the sludge was assumed to follow the solid fraction, and mixed with the drill cuttings the activity concentration in the ashes was expected to be in the range 2-6 Bq/g. The activity concentration in discharges to water and air were expected to be insignificant.

IFE was requested to carry out verification measurements during the incineration to ensure that the radioactivity levels were well within current regulations and within calculated levels for exhaust gases and water [8].

A sampling programme was set up covering the following sampling points: Coarse ash under incinerator, ash below the cooler outlet, ash from skimmer (auger), ash from main filter (auger), water from scrubber and offgas from stack. The external dose rate in the vicinity of the sampling points were measured using a Geiger counter of the type Automess AD5. In addition, contamination control, using an Electra contamination monitor, was performed along the transport route for the sludge. Two sampling rounds were performed the day before the incineration process began. The first day of incineration four sampling rounds were performed. A total of five visits were made during the incineration period, and two sampling rounds were performed at each visit. A final visit was made three weeks after the incineration period.

All samples were analysed using the same experimental set-up as described earlier. A selection of samples were analysed with high-resolution gamma spectrometry to establish a calibration factor for the measurements. At very low count rates, the uncertainty in the measurements would be substantial and a discrepancy between field and laboratory measurements was expected.

Analyses of the ash from the incineration showed activity concentrations well below the estimates given in [8], the highest concentration of ^{226}Ra being approximately 2.5 Bq/g. Samples of water and offgas showed activity concentrations below or close to the detection limit for gamma spectrometry. Dose rate measurements in working areas in the vicinity of the sampling points did not reveal any increase from the background level measured before the incineration began.

It was concluded that incineration of sludge with activity concentrations below the exemption level is a well-suited method for disposal. The dose to personnel at the incineration facility will under normal circumstances be very low. Risk for intake, especially inhalation of dust having elevated activity levels, should however be observed.

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Survey of Radioactivity in Scale from Demolition of an Oilrig

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An oilrig was decommissioned and brought to shore from the North Sea for demolition in June 2001. The Norwegian Radiation Protection Agency (NRPA) has published directions[1] on how deposits in pipes and tanks in oil and gas production (*scale*) should be controlled with respect to radioactivity.

Scale is formed by precipitation of sulphates and carbonates from the water in the production line in oil production. The naturally existing nuclides ^{226}Ra and ^{228}Ra are precipitated together with barium, calcium and strontium due to similar chemical properties. It is possible for the concentration of radium in scale to exceed the level in sedimentary rock.

Scale in the production of natural gas consists of thin layers of solid elements inside the production line. ^{210}Pb will concentrate in scale as it is a daughter of the inert gas ^{222}Rn, and has a longer half-life than the intervening nuclides.

The task

Aker Maritime is responsible for the demolition, and wants to send samples of scale to a laboratory for measurements. They contacted the Hospital and asked if we could do the measurements in our “low-activity” laboratory. At that time the laboratory needed an upgrade with new equipment and training of new personnel. We therefore started a co-operation with Laborel AS, who has supplied us with new measuring equipment and software. The laboratory now has NaI(Tl)- and HPGe-detectors with MCA and software for measurement control and spectral analysis.

Aker Maritime requires quick answers and prefers the laboratory to be located close to the demolition site. The Hospital fulfils these requirements. We

[1] StrålevernHefte 12, Avleiring av naturlig radioaktive stoffer i olje- og gassproduksjon, Statens strålevern 1997.

Also: StrålevernRapport 1999:2, Håndtering av radioaktive avleiringer i olje- og gassproduksjon i Norge, Storbritannia og Nederland, Statens strålevern og Oljedirektoratet 1999.

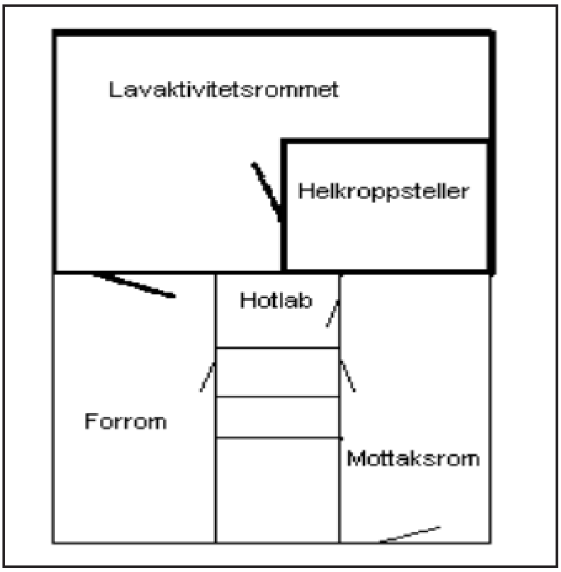

Figure 1.

can analyse samples with short measurement times partly because the "low-activity" laboratory has a good signal-to-noise ratio.

The directions given by the NRPA define requirements to the analysis of the samples. If the activity is more than 10 Bq/g for any of the isotopes ^{226}Ra, ^{228}Ra or ^{210}Pb, the deposits shall be handled specially.

Our results are given with an uncertainty of ±2 Bq/g. Low activity measurements are given the result <1 Bq/g.

"Low-activity" laboratory and methods

The entire laboratory (Lavaktivitetsrommet) is built from concrete and it is placed at the very bottom of the largest building at the Hospital (Figure 1). Inside the laboratory is another layer of shielding made of steel (Helkroppsteller). The inner compartment is large enough to contain a couch and the necessary equipment to perform whole-body measurements. All the building materials are specially selected to contain as little radioactivity as possible.

The laboratory is surveyed regularly to ensure that the rooms haven't been contaminated. The survey measurements are performed with a NaI-detector at various locations. From these measurements we have found that the counting rate inside the laboratory is approximately 20 times less than outside. The inner compartment reduces the measured count rate another 40 times. The scale measurements are performed in the outer laboratory, as the signal-to-noise ratio is sufficient low there. We are also very careful to avoid bringing possible sources of contamination inside the inner compartment.

The samples that we receive are packed in sample boxes and they are measured with a Geiger-Müller counter. We have compared these measurements with the measured activity of ^{226}Ra, and we have found fairly good agreement between counts per second and activity. While these results should be used with care, Geiger-Müller measurements give an opportunity to estimate the activity; and thus which sampling times will be necessary to achieve good statistics on the spectral analyses.

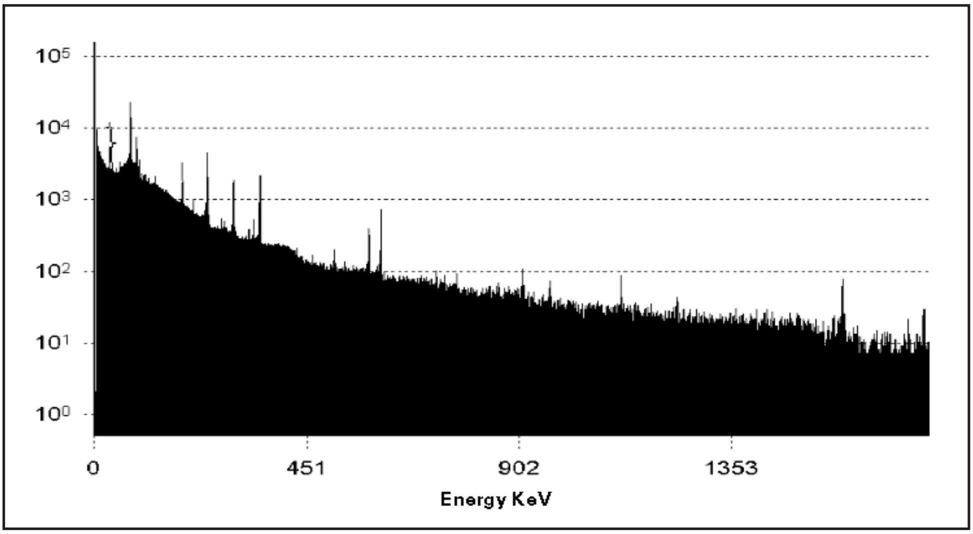

Figure 2.

The spectrum above (Figure 2) is measured with a HPGe semiconductor detector on a calibrated scale standard sample. These standards are used to calibrate the detector with respect to energy and efficiency. The software (Genie 2000) contains a system of calibration and certificate files that tells us when and how the detectors were last calibrated. We have found this useful in a multi-user environment.

The software handles the spectral analysis when the user has specified which nuclide library to use. The scale library contains the most prominent lines of the natural radioactive series as well as some nuclides that we may find in the samples (e.g. ^{40}K and ^{137}Cs).

^{226}Ra is measured by its 187 keV line. ^{235}U has several lines, one of which is 190 keV, and it could be difficult to separate the two. The activity of ^{235}U can be estimated from its other lines and subtracted. So far we haven't detected ^{235}U in the samples. ^{228}Ra will be in equilibrium with ^{228}Ac; which can readily be detected. ^{210}Pb can be measured directly by its 47 keV line.

Results

The initial samples were taken from storage tanks for oil and the measurements confirmed the expected results; <1 Bq/g. We have later received samples from the production line, and some of these have had radium activities above 10 Bq/g. So far we haven't measured activities of ^{210}Pb above the 10 Bq/g limit.

Three samples have been sent to STUK. The activities measured in two of the samples were in very good agreement, while the third deviated by 22%. This deviation is (according to STUK) probably due to density corrections and use of non-airtight sample boxes, which would let radon escape the box and upset equilibrium conditions.

Conclusion

This project has given us an opportunity to expand our activity into new areas in co-operation with commercial partners. The laboratory has been upgraded with new equipment and we have got valuable experience in measuring techniques. We now have better opportunities to offer new services in low-level radiation measurements. We consider offering surveys of personnel or patients who have been exposed to radioactivity.

The project also shows that the activity of the scale samples can readily be measured without using the inner compartment. We find that promising with respect to which activity levels it will be possible to detect inside the inner compartment.

Natural radionuclides in fuel and waste from Danish power plants

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Abstract
We have examined the content of natural occurring radionuclides in three types of fuel and waste from power plants in Eastern Denmark (coal, orimulsion and biomass). While coal previously was the main fuel used for power production, the majority of fuel used today is biomass, natural gas and orimulsion. Coal was in 2000 less than 50 %. It was found that the Ra-226 content of coal ash was on an average 104 Bq/kg, which is similar to what was found in previous studies. The Ra-226 content of orimulsion ash is only 8 Bq/kg, which exclude radiological problems with this fuel and the associated ash. Bio fly ash from combustion of straw contains about 40 % potassium and from a radiological point the K-40 content of up to 13000 Bq/kg is the most interesting. However, even handling of large amounts of bio fly ash will probably not give external doses to workers beyond the radiological protection goals of the European Basic Safety Standards.

Background and introduction
All types of coal, oil and other fossil fuel types contain small amounts of natural occurring radionuclides, principally K-40 and radionuclides from the U-238 and Th-232 decay series. Combustion of the fuel leads to an increased concentration of the nuclides in the ash. Several studies of the radiological impact on the population and on workers from coal-fired electricity generation have therefore been made [1,2,3].

A previous Danish investigation of coal and coal fly ashes from 1984, Ulbak [3] showed that the use of fly ash from coal-fired power plants in concrete increased the radiological impact of a habitant in a concrete apartment by 30 μSv/y.

For two reasons we have decided to reinvestigate the radiological impact from Danish power plants. Firstly, until 1995 the principle fuel in power plants in Denmark was coal (above 90 %). The majority of fuel used today is biomass (mainly straws), natural gas and orimulsion. Orimulsion is composed of 30% water and 70% natural bitumen. It is produced in Venezuela and is in Denmark only used at Asnæs Power Plant [4]. In 2000, coal amounted to less than 50 % of the fuel used for electricity generation. Secondly, in the case of natural occurring radioactive material (NORM), the European BSS [5] requires that the member states regulators identify which work activities and associated wastes or discharges require control or prior authorisation.

This study includes some of the power plants in Eastern Denmark organized in the company ENERGI E2, which produced both electricity and district heating on Sjælland, Eastern Denmark. Some of the studied power plants use several fuel types while other are dedicated to one fuel type (Table 1).

Material and analytical techniques

All samples were collected between January and July 2002. They were kept for one month in air-tight Marinelli beakers before they were analysed with a high-purity germanium detector. Ra-226 was determined by the gamma lines of its progenies Pb-214 at 353 keV and Bi-214 at 609 keV, 1120 keV and 1764 keV assuming secular isotopic equilibrium with Ra-226. Th-232 was determined from the gamma lines of it progenies Tl-208 and Ac-228 at 583 keV and 911 keV respectively, also assuming equilibrium with Th-232. K-40 was determined directly via the 1461 keV gamma line. All samples were analysed for a minimum of 60000 s. Errors for the Ra ,Th and K activity concentrations are ± 20% or less and include counting statistics, determination of the detector efficiency, and correction for photon mass attenuation.

Results

The results of the analyses are shown in Tabel 1. Biomass is in Denmark mostly straw, which comes from farming in Eastern Denmark. For all fuel types, both the unburned fuel and the remains (ash) have been analysed. A portion of the ash remains in the boilers after combustion and is termed bottom ash. The smaller and lighter ash particles, which escape the boiler and are captured in a filter are termed fly ash. The ash content of the various fuel types depends both on the quality of the coal or biomass and on the type of boiler. The fly ash

Table 1. Activity ratios of natural radionuclides of samples from Danish power plants. Bio mass is straw if nothing else is given. Errors for activity concentrations are ± 20% or less.

Sample no.	Sample type	Power plant	K-40 (Bq/kg)	Ra-226 (Bq/kg)	Th-232 (Bq/kg)
1	**Coal**	Avedøre	71	15	14
8		Asnæs	77	9	6
9			78	10	6
14			57	11	8
15			65	11	10
2	**Coal fly ash**	Avedøre	669	158	103
4			693	146	111
6		Asnæs	292	88	79
7			393	87	78
12			562	105	84
13			554	102	83
37	**Coal bottom ash**	Amagerværket	581	102	83
3		Avedøre	396	83	63
5			453	85	70
10		Asnæs	371	59	49
11			361	62	54
18	**Orimulsion**	Asnæs	0	0	0
19			0	0	1
16	**Orimulsion ash**		25	8	28
17			26	8	27
30	**Bio fly ash**	Avedøre	12804	9	3
32			12070	10	5
33			12389	11	5
20		Slagelse	11902	13	7
21			11839	12	6
28		Masnedø *	11531	16	10
29			11703	18	8
31	**Bio bottom ash**	Avedøre	1940	42	14
22		Ringsted**	1196	10	4
23			654	6	3
24		Haslev**	3631	34	16
25			3535	35	17
26			7231	24	11
27			8130	25	12

* Ash from fuel consisting of about 20 % wood and 80 % straw.

** The samples are a mix of fly ash and bottom ash.

content of coal is typical between 10 and 20 % while it is only between 0,5 and 1 % for biomass.

Discussion

The coal and coal ash samples analysed in this study have activity concentrations similar to what is found in previous studies [2, 3]. Coal fly ash is used in the same way as for many years in the production of concrete, cement and asphalt. As previously mentioned Ulbak [3] has calculated the radiological impact of use of coal fly ash in concrete and these calculations are still valid today.

Orimulsion is a relatively new fuel in Denmark but the very low content of naturally occurring radionuclides in this fuel and in the ash excludes any radiological impact for workers handling orimulsion. The Ra-226 content of approximately 8 Bq/kg is much lower than in almost any type of soil or rock in Denmark [6]. The orimulsion ash is exported to companies in the UK, Germany and Austria, which exploit its content of metals.

The analyses of the bio fly ash samples reflect the large amount of potassium in these samples (about 40 %). Bio fly ash therefore also has a relatively high content of K-40 (up to 13000 Bq/kg). While fly ash from coal combustion in readily used in building material, most bio fly ash cannot be reused due to its high content of cadmium. This means that the power plants at the present store the ash in 'big bags' containing about 1 m^3 of ash each.

In order to meet the Danish parliament's targets for use of biomass, ENERGI E2 has initiated a comprehensive plan for the expansion of biomass-fired production, initially based on straw [4]. Over the coming years, E2 expects to use 4 times as much biomass fuel as they do today, which means that they will burn about 350 000 tons of straw a year. The newly build boiler at Avedøre alone has a capacity of 150 000 tons of straw each year making this the larges biomass-fired boiler in the world. Since the fly ash content of straw is only about 0,5 to 1 % of the original mass, they will produce a minimum of 750 tons of bio fly ash each year with a total activity of about 9,6 GBq K-40. Since the density of bio fly ash is about 200 kg/m^3 the total volume of ash will be on approximately 3750 m^3. This amount of ash will require some handling and possible disposal at a landfill site, which could potentially lead to doses to workers.

In compliance with the European BSS a dose criterion of 0.3 mSv/y (after background subtraction) have been used to calculate exemption/clearance levels for K-40 for various working scenarios [7]. A scenario where this dose rate could be reached describes exposure situations for workers at a NORM landfill site. For external gamma radiation, an exposure geometry of a point situated one metre above a semi-infinite volume is used and the worker is shielded by a steel vehicle. It is assumed that a worker spends 1800 h/a working with handling NORM, that the activity of K-40 on an average is 10 Bq/g, and that the disposal rate is at a minimum 20 tons/h or 36,000 tons a year. It is clear that compared to a future work scenario at Avedøre, or even the combined straw-fired power production at E2, it will be many years before handling of NORM will amount to a volume that requires dose control.

Conclusions

The development in power production in Denmark means that the potential radiological impact from natural radionuclides in fuel and fuel waste is currently undergoing changes. The most important change is the wide use of biomass (straw or wood), which is replacing coal as the major fuel. The potential radiological hazard from bio ash is external radiation from K-40. This ash type cannot at the present be reused and is therefore stored in large quantities. Preliminary calculations indicate that external radiation to workers is not likely to exceed 0,3 mSv/y even with a much larger production than today. However, since much more biomass will be used in the near future, it is relevant for regulators to monitor this development and to make actual monitoring of doses to workers and the public.

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^{210}Pb in Air and Soil: Ra -Rich Areas in NE Estonia

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The present report presents the results of a detailed study of the ^{210}Pb and ^{226}Ra in undisturbed soil profiles collected from Lääne-Virumaa and Ida-Virumaa County in NE Estonia. In a few areas of this region, an enhanced natural U/Ra content characterizes surface soils. In general, maximum U/Ra activity concentrations found in these distinct areas increase from west to east. These Ra-rich areas are neighboring with areas characterized by a significantly lower Ra con-tent. From place to place the range from 25 Bq kg^{-1} to 350 Bq kg^{-1} has been determined [1]. At present, data on the detailed geographical distribution of these areas are rather scarce.

Depth-distributions and inventories of the ^{210}Pb activity have been determined from the HPGe gamma spectrometric analysis of the 2-3 cm thick slices of soil profiles. In spectrum analysis [2], a self-attenuation correction method basing on a direct 46.5 keV transmission measurement for each sample was applied. In addition, for the same samples placed into sealed beakers ^{226}Ra content and ^{222}Rn emanation coefficients have also been determined by using gamma spectrometry.

Along a typical soil profile, the ^{226}Ra concentration showed a negligible variation, while the activity concentration of ^{210}Pb demonstrates a considerably more complex behavior with depth. In upper soil layers two ^{210}Pb fractions exist: the atmospherically deposited unsupported fraction and the in-situ produced supported fraction fixed to the soil particles. An attempt has been made to separate the fractions as a function of depth.

From the determined unsupported fraction inventories in low-Ra soils, ^{210}Pb deposition fluxes and concentrations in precipitation have been evaluated. In these cases, a depth-dependent activity concentration of the unsupported fraction has been reasonably well approximated by an expo-nential (and for a few cases, by a lognormal) distribution.

In Ra-rich soils, which serve as efficient sources of radon, the unsupported Pb fraction is comparable to or even smaller than the supported fraction and an appropriate modeling is required for separation of the fractions. A simple one-dimensional diffusion model has been satisfactorily applied. While mean deposition flux value, F, equals to 116 Bq m^{-2} a^{-1}, the maximum of 177 Bq m^{-2} a^{-1} has been determined. The concentration in precipitation reaches values

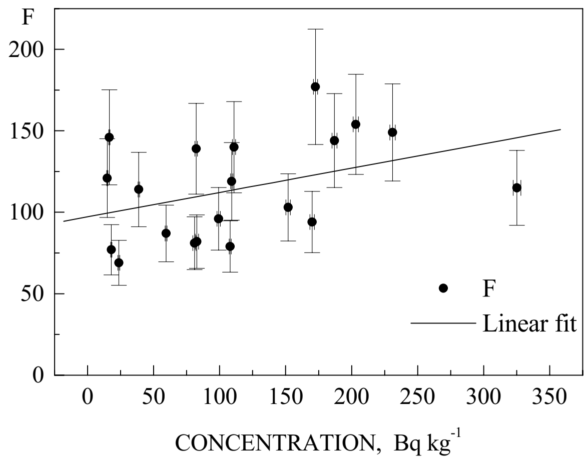

Figure 1. ^{210}Pb deposition flux, F, vs ^{226}Ra activity concentration, C_{Ra}, in soil, 19 sites in NE Estonia, 1999-2001. Solid line is a linear fit, $F=97+0.15\ C_{Ra}$

up to 260 Bq m^{-3}, using a local long-term annual mean precipitation value of 0.68 m a^{-1}. The deposition flux shows a tendency for finding higher deposition fluxes at the locations characterized by higher ^{226}Ra concentration, C_{Ra}, in soil. The relevant dependence F vs C_{Ra} with a linear fit is illustrated in Figure 1, which shows a slight increase in flux, while the Ra concentration increases. The correlation coefficient is 0.39.

The found correlation may be explained assuming locally enhanced ^{210}Pb (and ^{222}Rn) concentrations in outdoor air of the Ra-rich NE regions in comparison to the worldwide reference value of 0.5 mBq m^3 [3].

Preliminary results on the low-energy gamma spectrometric analysis of ^{210}Pb in filter samples collected by high-volume air samplers in three different locations, Harku, Tõravere and Narva-Jõesuu, are obtained. The average and extreme ^{210}Pb concentrations in air (mBq m^{-3}) of these locations are presented in Table 1.

The presented air concentration data seem to confirm the above assumption about locally enhanced ^{210}Pb air concentrations in this region. The mean concentration value for the seaside station in Narva-Jõesuu (NE Estonia), neighboring the high radium areas, is almost equal to that for the inland station in Tõravere (South Estonia). Additional evidence follows also from the results of a transport modelling study for the airborne ^{210}Pb [4], where Eastern Estonia is identified among the ^{210}Pb source regions of a relatively high

Table 1. ^{210}Pb concentrations in air for three locations in Estonia, 2001.

	Concentration in air (mBq m^{-3})		
	HARKU	N-JÕESUU	TÕRAVERE
Arithm. Mean	0.47	0.57	0.55
Geom. Mean	0.37	0.44	0.46
Standard Error	0.08	0.09	0.08
Min - max	0.08 – 1.30	0.12 – 1.60	0.16 – 1.51

intensity.

The authors acknowledge the Estonian Radiation Protection Centre for presenting air filter samples and the ESF for a partial financial support.

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Posters

3a Continuous Monitoring of Indoor Radon

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A recent radon survey performed by one of the authors (L.P.) using passive plastic track detectors has confirmed that the time-integral air concentrations in living apartments are as an average about 100 Bq m^{-3} in Tartu and Tartumaa County. The average value for this region almost coincides with the corresponding country mean of 101 Bq m^{-3} [1]. In the region, the maximum average long-term radon concentrations up to 220 Bq m^{-3} have been found in a multi-storey mansion. It is well known that short-term radon concentration values may vary significantly from a long-term average.

The present report presents data on continuous radon monitoring carried out using an active alpha-spectrometric monitor ALNOR RM-3B in a work place during 2 years. The monitor was installed in a laboratory room of the Institute of Physics, University of Tartu, in the third floor of the five-storey building (58°21' N, 26°41' E). The building constructed in the 1970's has no air-conditioning and its ventilation is rather poor. The room accommodates two low background HPGe gamma spectrometers.

Radon was let to diffuse freely into the measuring chamber of the monitor, where both ^{218}Po and ^{214}Po radon daughters were counted, integrating over 4 hr (slow mode). In parallel, for characterization of the environmental conditions in the analysis room, gamma dose rate, room temperature, relative humidity and atmospheric pressure were also continuously recorded.

During the measurement period, daily radon concentration varied in the range from 8 Bq m^{-3} to 71 Bqm^{-3} with a long-term average value of 29 Bq m^{-3} and a standard deviation of 9.6 Bq m^{-3} (Fig. 1). At the same time minimum and maximum values of 4 h measurements vary in the range from 5 to 80 Bq m^{-3}. Distinct day-night, weekly and yearly oscillations in air concentration of radon have been found (Fig. 2).

The measured data are analyzed for correlation between radon concentration and the environmental parameters (temperature, humidity, etc.). The analysis is in progress and only the most apparent correlations are available. E.g., in addition to the above-mentioned time-dependent features, a significant inverse correlation with air pressure is observed.

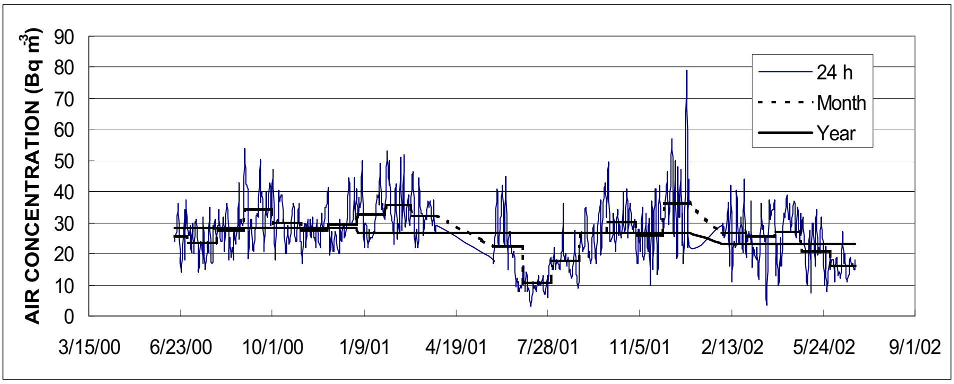

Fig. 1. Radon concentration in laboratory air averaged over: 24 h, 1 month and 1 year.

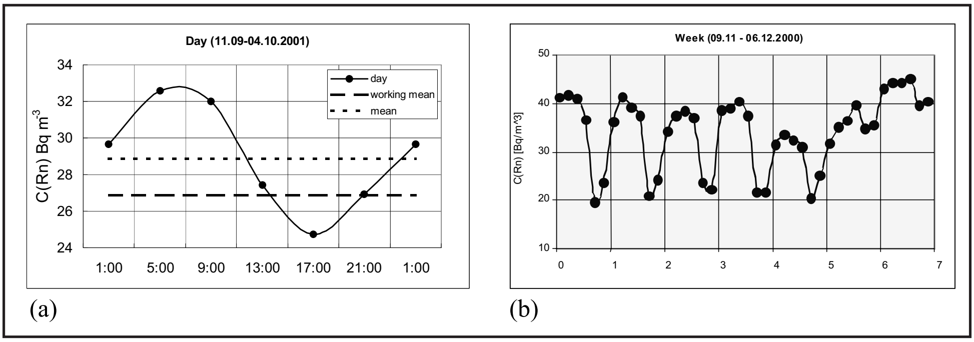

Fig. 2. Average (a) daily and (b) weekly dependence of radon concentration in room air. Overall mean and mean calculated for working hours are also shown.

The results are discussed in relation to the exposure of persons and to the low-level counting. The present results are also a part of monitoring carried out in the framework of the local quality assurance program for low-background gamma spectrometric analysis.

The authors acknowledge a partial support from the ESF grants.

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3b NORM Exposure to Workers at a Zircon Sand Processing Factory

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Abstract
The National Institute of Radiation Hygiene (NIRH) has estimated doses to workers at a Danish zircon-processing industry. Calculations of internal doses were made after measuring dust concentrations in the working area and content of radium-226 and thorium-232 in the zircon sand. The total effective dose from internal and external irradiation was calculated to approximately 0,75 mSv/y.

Introduction
Zircon ($ZrSiO_4$) is a natural occurring mineral with a relatively high content of radionuclides from the uranium and thorium decay series. Activity concentrations for zircon sand are reported to be in the range from 0,4 Bq/g to 40 Bq/g of thorium-232 and from 0,2 Bq/g to 74 Bq/g of uranium-238 [1].

Because of its high melting point, zircon is used in the manufacturing of refractory materials. It is used in the industry as a fine powder, and workers can be exposed to NORM internally by ingestion, by inhalation of dust, by skin contamination and from inhalation of radon. External NORM exposure can occur in the proximity of large quantities of material such as in a storehouse.

Historically, mining and processing of zircon sand have lead to high exposures from contaminated dust, although in the last ten years much has been done by the industry to reduce such exposures. Typical dust concentration levels in refractory fabrication plants are between 0,7 and 3,1 mg/m^3 [1].

NIRH has calculated the annual effective dose to workers at a Danish zircon sand processing factory. Calculations of internal doses were made after measuring dust concentrations in the working area and content of radium-226 and thorium-232 in the zircon sand.

Materials and Methods

Samples of three different types of zircon sand have been analysed for the concentration of radionuclides. The samples were analysed in the Dosimetry Laboratory at NIRH using a gamma spectrometer with a Canberra High Purity Germanium detector, Model GC 2318-7500. The samples were kept for one month in airtight Marinelli beakers before they were analysed for a minimum of 60000 s. Radionuclides were determined following the procedure described in [2].

Dust particles in the air were collected and separated in size fractions by means of an impactor. The impactor collects particles in 5 size fractions between 0,36 µm and 6 µm. The dust particles are deposited on a glass plate coated with apiezon. A pump is used to force the air through the impactor with a flow rate on 15 L/minute [3].

Dose rate from external irradiation was measured with a Bicron microsievert instrument.

Results

Concentrations of radium-226 and thorium-232 in three different types of zircon sand are shown in table 1. The uncertainty is approximately ± 20 %.

Table 1. Activity concentrations in zircon sand, AC_{SAND}.

Sample	AC_{SAND} (Ra-226)	AC_{SAND} (Th-232)
1	3,1 (Bq/g)	0,57 (Bq/g)
2	4,3 (Bq/g)	0,78 (Bq/g)
3	3,0 (Bq/g)	0,67 (Bq/g)

Dust concentrations were measured at the working area under the most dusty conditions, where zircon sand in big bags are poured into the production equipment. In this experiment 437 L of air was pumped through the impactor. The concentration of dust in the place of production was higher than normal, because the ventilation in the production equipment was wrongly adjusted. The measurements reported in table 2 therefore describe an unlikely scenario with dust concentrations higher than normal.

Figure 1. Impactor.

Table 2. Mass of dust from each impactor step.

Impactor step	1	2	3	4	5	Filter paper
Particle Size [3]	> 6 μm	6-2,5 μm	2,5-1,7 μm	1,7-0,70 μm	0,70-0,36 μm	< 0,36 μm
Mass	7,4 mg	5,9 mg	2,1 mg	1,5 mg	0,70 mg	0,95 mg

Dust particles from impactor step 1 were excluded, because these particles are larger than 5 μm and cannot be inhaled into the alveolar of the lungs [4]. The concentration of dust in the air that can be inhaled, C_{DUST}, was calculated to 25 mg per m^3.

The dose rate from external irradiation in the place of production was measured to approximately 0,15 μSv/h.

Discussion

Because the activity concentrations of the zircon sand were higher than the exemption levels for naturally occurring radionuclides [5,6], it was decided to estimate the annual effective doses to the workers.

The effective dose from each radionuclide was calculated separately for the five exposure pathways. Doses from all radionuclides were summed to give the total dose for each pathway. Finally, the doses from the pathways were summed to give the total effective dose. It was assumed that uranium-238, uranium-235 and thorium-232 were present in secular isotopic equilibrium with all their daughter products. The calculation of doses from each pathway is described in [1].

Dose from inhalation of dust from each radionuclide is given by the following equation [1].

Inhalation of dust:

• $UD_{INH} = DC_{INH} * R_{INH} * C_{DUST} * T_{INH}$
• Where UD_{INH} = The annual effective dose per unit activity from inhaling contaminated dust, in Sv y^{-1} per Bq g^{-1}
• DC_{INH} = Dose coefficient for inhalation, in Sv Bq^{-1}
• R_{INH} = The rate at which air containing contaminated dust is breathed by the individual, in m^3 h^{-1}
• C_{DUST} = The concentration of contaminated dust in the air, in g m^{-3}
• T_{INH} = The time that an individual is exposed to these conditions, in h y^{-1}

Table 3 shows the parameters used to calculate the annual effective dose for the unlikely working situation. It is assumed, that the activity concentrations in the size fractions are the same as in the bulk material. No respiratory protection was used.

Table 3. Parameters for inhalation of dust. AC_{SAND} is the activity concentrations in Bq/g from table 1. E_{INH} is the effective dose from inhalation in mSv/y.

Decay series	DC_{INH}	R_{INH}	C_{DUST}	T_{INH}	UD_{INH}	AC_{SAND}	E_{INH}
U-238 + daughters	6,8E-05	1,18	2,5E-02	10	1,9E-02	3,5	0,068
U-235 + daughters	7,0E-04	1,18	2,5E-02	10	2,1E-01	0,16	0,033
Th-232 + daughters	7,6E-05	1,18	2,5E-02	10	2,2E-02	0,67	0,016

The total effective dose, E_{INH}, from inhalation of dust using the parameter values in table 3 is 0,12 mSv/y.

Under normal conditions the concentration of inhaled dust C_{DUST} was estimated to 1 mg per m^3, and the time an individual is exposed T_{INH} was estimated to 1000 hours/y. The total internal effective dose is then calculated to 0,47 mSv/y. The external dose at the working area is 0,15 mSv/y, whereas doses from ingestion and skin contamination are negligible. Because the factory does not store large amounts of zircon sand, doses from inhalation of radon were not calculated. The total effective dose from internal and external irradiation was thus estimated to approximately 0,75 mSv/y.

Conclusion

According to Danish legislation [6], industries that process material with enhanced levels of naturally occurring radionuclides require an appropriate form of regulatory control, if the annual effective dose to workers or members of the public exceeds 0,3 mSv/y.

The calculated annual effective dose of 0,75 mSv/y for this factory indicates, that the zircon processing industry should be further investigated for the radiological impact from natural radionuclides. The factory will also require a license from the authorities, because the activity concentrations of uranium-238 in the zircon sand were higher than the exemption levels for this nuclide [6].

For further investigation it could be interesting to locate refractory producing factories, because refractory products are finished by grinding and polishing. One of the most radiological hazardous areas of a typical refractory producing factory is the sprue elimination area, where the castings are cleaned up [1]. In this area the concentration of dust is high, and the ventilation and dust control have to be adequate.

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3c Assessment of human exposure to natural radionuclides by using *in vivo* measurements

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Introduction

Geologically, Finland is part of the Fennoscandian shield and the bedrock consists mainly of granitoids. The average concentrations of natural radionuclides in granitoids, especially in granites, are high. Consequently, exceptionally high concentrations of natural uranium ($^{234,235,238}U$) and the daughter nuclides of uranium and thorium series (^{222}Rn, $^{226,228}Ra$, ^{210}Po, ^{210}Pb) have been found in wells drilled in bedrock. The concentrations of natural radionuclides both in drinking water and indoor air have been observed to vary markedly in Finland depending on the location. The highest concentrations have been measured in the granite areas of Southern Finland [1-3].

Exposure to natural radionuclides in drinking water is usually chronic and may persist for many years before it is recognised. It is normally difficult to assess individual cumulative exposures of afflicted persons with biokinetic models because of several problems: activity concentrations in water may show great temporal variations, assumptions on individual water consumption rates have to be made and many biokinetic parameters are uncertain. Due to these difficulties with biokinetic models it would be useful to have a biological indicator in the human body itself. This would provide more direct information on the cumulative exposure. The objective of this study is to demonstrate the feasibility of estimating individuals' cumulative exposure to uranium and ^{210}Pb by measuring the amounts of these nuclides in their skeletons.

Experimental methods

The short-lived decay products of inhaled ^{222}Rn deposited in the respiratory track decay inside the body into ^{210}Pb. Lead accumulates mainly (55%) in the skeleton and is retained there with long effective half-life. Several researchers have shown [4-6] that the human skull is suitable for *in vivo* measurements of ^{210}Pb in the human skeleton, whereas studies on the individual exposure via waterborne natural radionuclides are very scarce [7]. In this paper we have studied persons whose main source of long-lived radon progeny ^{210}Pb (22.3-year physical half-life) and natural uranium is drinking water.

Both lead and uranium accumulate mainly in the skeleton. Fractions of 0.6, 0.2 and 0.2 of the lead translocated to the skeleton are retained with biological half-lives of 12, 180 and 10000 d, respectively [8]. From uranium, fractions of 0.2 and 0.023 are transferred to mineral bone and retained there with half-lives of 20 and 5000 d, respectively [9]. The skull represents a fraction of 13-15% of fresh skeletal weight and anatomical structures of individuals' skulls do not differ much from each other. Additionally, a relatively thin layer of tissue around the skull makes it very suitable for uranium and ^{210}Pb measurements.

^{210}Pb is a beta emitter but 4.25% of its decay is followed by the emission of a 46.5 keV γ-ray. Of natural uranium isotopes ($^{234,235,238}U$) only the decay of ^{235}U is accompanied by the emission of γ-rays (186, 144, 163 and 205 keV) with intensities high enough (57, 11, 5 and 5%) to be detected *in vivo* using HPGe detectors. As no fluctuations in the isotopic abundance ratio of ^{238}U and ^{235}U in naturally occurring materials have been found, the concentration of ^{238}U can be calculated from the ^{235}U concentration [10]. Furthermore, it is possible to determine the ^{238}U content using the 63 keV and 93 keV peaks originating from the decay of its daughter nuclide, ^{234}Th.

The instrumentation constructed at STUK for the *in vivo* measurements consists of four high purity broad energy Ge-detectors: three detectors with crystal thickness of 20 mm and diameter of 70 mm and one detector with thickness of 20 mm and diameter of 80 mm. In part of the measurements one of the smaller detectors was replaced with a LOAX detector with a crystal diameter of 70 mm and thickness of 30 mm. Detectors are placed near the top and the back of the head, and lateral surfaces (see Figure 1). The average distance between the detector and head surface is approximately 10 mm. To reduce the amount of background radiation from the surroundings the instrumentation has been installed into a room with 15-mm thick iron walls covered with 3-mm lead and 2-mm copper. A measuring time of 7200 seconds was used.

Figure 1. Array of four broad energy Ge-detectors installed for *in vivo* measurements of natural radionuclides accumulated in the skull.

The study persons were selected from the drinking water database of STUK, which covers the results of radionuclide measurements from more than 7 000 drilled wells. Those individuals who had used water with highest ^{210}Pb and uranium concentrations were mainly contacted. By the end of June 2002 the levels of ^{210}Pb and uranium in the skull of thirteen Finns had been determined. A questionnaire was made to collect information on residential history, usage of drilled well water, daily beverage consumption (well water and other beverages) as well as smoking history and dietary habits. The study persons were also asked to collect a 24-h urine samples for the determination of ^{210}Pb and ^{nat}U concentrations. Uranium in urine (mg/l) was analysed with inductively coupled plasma mass spectrometry (ICP-MS) in the laboratory of Consulting Engineers Paavo Ristola Laboratory, Hollola, Finland. The analysis method for ^{210}Pb in urine is under development.

Results

Each study person was measured for 7 200 seconds in periods of 1 800 seconds. Figure 2 shows a few examples of collected γ-ray energy spectra. Spectra a) and b) were recorded from subjects who use water rich in uranium and ^{210}Pb, whereas spectrum c) represents a control person exposed to normal ^{210}Pb and ^{nat}U concentrations in the environment. Examples a) and b) are members of the same family (a daughter and father) who have been using the same water source for almost 20 years. Enhanced ^{210}Pb and ^{235}U activities were observed in both. However, much smaller amounts of uranium and ^{210}Pb have accumulated in the father. According to the questionnaire the father reduced his water consumption from the family's drilled well about 15 years ago. In addition, he has consumed the water only as an adult whereas the daughter consumed the same water during her growth period. The results of the measurements (water, urine and skull) are shown in Table 1. The radionuclide concentrations of the well water have been measured several times in different years. Some of the families bought water treatment equipment for removing uranium or radon from water as soon as the enhanced radionuclide concentrations were noticed. In the table, the higher concentrations represent the situation when no water treatment equipment has been used. The seasonal variations in radionuclide concentrations have been large in some of the drilled wells used by the study individuals. The results related to the Figure 1 are represented in the Table 1. The background (the lowest row) has been subtracted from the results of the skull measurements.

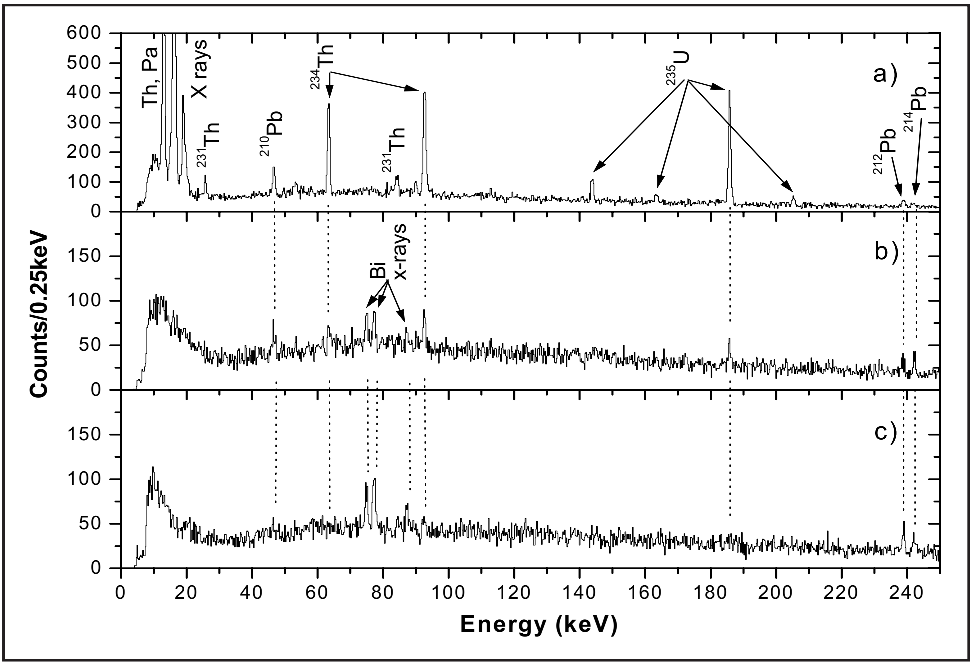

Figure 2. a,b) Gamma-ray energy spectra recorded from subjects who use water rich in natural uranium and ^{210}Pb. Spectrum c) was recorded from a control person exposed to normal ^{210}Pb and ^{nat}U concentrations in the environment. The background in the spectra may vary due to different measurement dates. Spectrum a) was recorded in June 2002 while the others were collected in March 2002.

Discussion

Although the number of studied individuals is too small to be statistically significant, a positive correlation can be observed between the U concentration in urine (µg/d) and the number of counts (cpm) in the 186 keV and 63 keV peaks originating from the decay of ^{235}U and ^{234}Th, respectively. Likewise a correlation can also be seen between the estimated intakes of uranium (mg) from drinking water and the counts in the 186 keV and 63 keV peaks. The estimation of the total intake of uranium in the course of years was based on the uranium concentration in drilled well water and on the information of the water consumption obtained by the questionnaire. A rough estimation of the uranium intakes was possible after the time of the installation of water treatment equipment and the variation in activity concentrations in water (see Table 1) were known. At the moment, the absolute efficiency of the detection system is not known due to lack of a realistic anthropomorphic skull phantom. After the calibration it will be possible to compare the results of the *in vivo* measurements to those obtained from the activity concentrations in water and in urine using biokinetic models.

Drinking water is the predominant source of uranium and ^{210}Pb in our study, but the contribution of each pathway for uptake into the human body have to be considered. Unfortunately, there is no data on uranium and lead contents in normal Finnish diet. The recent studies performed elsewhere show that the average daily uptake of ^{210}Pb in a ‘normal’ indoor atmosphere (C_{Rn}=50 Bq/m^3, F=0.4, C_{Pb}=0.5 mBq/m^3 [11]) is 23 mBq [6] and references therein. The contribution of atmospheric ^{210}Pb, domestic radon and diet to ^{210}Pb uptake is 12%, 2% and 86%, respectively. Tap water with low or normal concentration of ^{210}Pb has been estimated to be a negligible source of ^{210}Pb [12], while smoking a pack of 20 cigarettes would result in an additional ^{210}Pb uptake of about 10 mBq/d. The average daily intake of uranium from foodstuff has been reported to be 1-2 µg/d [11]. Small amounts of uranium and ^{210}Pb may also accumulate in the hair. To estimate the effect of this source of background, hair samples from study persons will be collected and analysed.

This study indicates that the measurement of the ^{210}Pb and ^{nat}U contents in the skull could be applied to estimate the individual cumulative exposures to the waterborne activities. The calibration of the measurement set-up and the determination of background sources are in progress.

Table 1. The results of the measurements.

study person	Radionuclides in water ^{nat}U µg/l kBq/l	^{222}Rn	Years of water usage	Years of water treatment in operation	Urine U µg/d	Radionuclides in skull cpm ^{234}Th 63 keV	^{235}U 186 keV	^{210}Pb 46 keV
male	50-12500	1-9	6	-	8.11	6.3±0.2	5.0 ±0.2	2.4 ± 0.1
female	50-12500	1-9	6	-	12.17	18.4±0.4	12.2± 0.3	5.2 ± 0.2
male	1-30	0.1-27	19	6 (Rn)	0.28	-	0.3 ± 0.1	0.5 ± 0.1
female	1-30	0.1-27	19	6 (Rn)	0.33	0.3±0.1	0.8 ± 0.1	1.6 ± 0.1
male	1-30	0.1-27	19	6 (Rn)	-	0.2±0.1	0.7 ± 0.1	1.9 ± 0.1
female*	1-30	0.1-27	10	6 (Rn)	-	0.1±0.03	0.7 ± 0.1	1.8 ± 0.1
male	3800-6300 **a)**	9-12	22	-	1.10	0.8±0.1	0.4 ± 0.1	0.5 ± 0.1
female	3800-6300	9-12	22	-	16.71	5.5±0.2	4.1 ± 0.2	2.4 ± 0.1
female	3800-6300 **b)**	9-12	19	-		9.4±0.3	8.2 ± 0.3	3.4 ± 0.2
male	1-100	0.1-25	13	7 (Rn), 3 (U)	0.06	0.5±0.1	0.1± 0.03	0.1 ± 0.03
male		0.1-38	16	15 (U), 3 (Rn)		-	0.2 ± 0.1	0.5 ± 0.1
male		6-7	17	3 (U&Rn)		0.3±0.1	0.2 ± 0.1	0.4 ± 0.1
female		6-7	17	3 (U&Rn)		0.3±0.1	0.3 ± 0.1	0.7 ± 0.1
back ground						0.2	0.3-0.5	0.2-0.4

*Child
a,b) See Figure 2.

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3d Natural radionuclides and ^{90}Sr in sludge at waterworks and wastewater treatment plants

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Introduction

Many radionuclides in water are concentrated into sludge during water treatment process. Sewage sludge is a sensitive indicator of radionuclides entering the environment. In previous studies artificial gamma-emitting radionuclides from the Chernobyl accident were found in activity measurements of sludge from waterworks and wastewater treatment plants. There were also found radionuclides from nuclear weapon tests, nuclear power plants and medical use in sewage sludge. These results have been reported in STUK report series[1-4]. To obtain more comprehensive information on radionuclides in sludge, the amounts of natural radionuclides and ^{90}Sr were determined in sludge samples taken earlier from waterworks and wastewater treatment plants. Natural radionuclides were principally analysed from the samples taken in 1991 and 2001, and ^{90}Sr from the samples taken in 1986, 1994 and 2001.

The investigation determining the activity concentrations of natural radionuclides in sludge gives more information on their contribution to radiation doses of workers at the wastewater treatment plants. The investigation also gives knowledge about the transfer of radionuclides in environment and possible radioecological consequences when using sewage sludge as fertiliser or as a soil-improvement agent in the fields. In addition, the study provides information for incineration of sludge.

Waterworks

Data on plants and treatment methods are given in Table 1. Sludge samples formed in the surface water treatment plants during water purification process were obtained from the treatment plants Pitkäkoski in Helsinki and Dämman in Espoo. Pitkäkoski receives raw water via a 120-km long rock tunnel from Lake Päijänne. The transportation of water in the tunnel takes about ten days.

Waterworks	Treatment method	Year	Raw water from lake/river	Sludge to wastewater treatment plant
Helsinki, Pitkäkoski	Al-sulph	1991	Päijänne	Kyläsaari
Helsinki, Pitkäkoski	Fe-sulph	2001	Päijänne	Viikinmäki
Helsinki, Pitkäkoski	Fe-sulph	2001	Vantaanjoki	Viikinmäki
Espoo, Dämman	Al-sulph	1991	Dämman	Suomenoja
Espoo, Dämman	Al, Fe	2001	Dämman	Suomenoja

Wastewater treatment plant	Treatment method	Dry matter %	Estimated amount of sludge $m^3\ d^{-1}$
Helsinki, Kyläsaari	Biological-chemical simultaneous precipitation (RS)	16 - 22	130
Helsinki, Viikinmäki	RS	30 - 33	200
Espoo, Suomenoja	RS	c.30	55 - 70
Rauma, Maanpääniemi	RS	12 - 14	25 - 30
Pori ,Luotsinmäki	Chemical treatment	15 - 18	50 - 60
Kotka, Mussalo	RS	16 - 18	c. 23
Kotka, Sunila	RS	15 - 20	20 - 30
Loviisa, Vårdö	Direct precipitation	18 - 20	3 - 4

Table 1. Treatment methods of waterworks and wastewater treatment plants and mean values for dry matter and estimated amounts of sewage sludge during sampling.

Water in waterworks includes about 10% of groundwater, which infiltrates into the tunnel from the bedrock. In the city of Espoo about 70% of water comes from Lake Päijänne via Pitkäkoski waterworks and 30% from Lake Dämman. Pitkäkoski plant treated water using aluminium sulphate until 2000. New methods using iron compounds have been adapted since 2000. The tunnel from Lake Päijänne was closed from September 1,2001 to December 22, 2001 and the Pitkäkoski plant received water from the Vantaanjoki river. In Dämman plant aluminium sulphate was changed to polyaluminium chloride at the beginning of 1990's and again to iron chloride in 2000. The older sludge sample taken from Dämman in 2001 could contain Al-sludge.

The sludge from the Pitkäkoski water treatment plant in Helsinki is pumped into the sewer system leading to Viikinmäki wastewater treatment plant (until 1994 to Kyläsaari plant) and the sludge from Dämman to Suomenoja wastewater treatment plant.

Wastewater treatment plants

The sludge samples were taken in municipal wastewater treatment plants from the output sludge of the plant and also in Kyläsaari (to 1994) and Viikinmäki (since 1994) in Helsinki from the raw sludge before digestion and dewatering. Data on the wastewater treatment plants are given in Table 1. In the communities studied, groundwater was used for tap water in the cities of Loviisa and Pori, and surface water in other places.

Methods

In our laboratory the digested, dewatered sewage sludges have been gammaspectrometrically measured in Marinelli beakers, without any pretreatment. Other sludge samples have been dried at a temperature of 105 °C, homogenised and measured using a plastic beaker. The measurement time was about 1000 min.

Separation and measurement of ^{90}Sr

The method of analysis was based on the radiochemical separation of ^{90}Sr through oxalate, nitrate, chromate and carbonate precipitation (Bryant et al 1959, modified by STUK). ^{90}Sr and ^{90}Y were determined together after ingrowth of ^{90}Y using Lumagel liquid scintillation cocktail and the Quantulus Liquid Scintillation counter. The measuring time was 300 min and efficiency of ^{90}Sr was 199.6%

Gammaspectrometric measurements

Artificial gamma-emitting radionuclides were earlier measured gammaspectrometrically from samples taken from waterworks and wastewater treatment plants. The same gammaspectra were now used for evaluating the amounts of natural radionuclides in these samples.

^{228}Ra was determined from the short-lived daughter ^{228}Ac. ^{226}Ra was determined from the 186.1 keV line taking into account the interference of the 185.7 keV line of ^{235}U. ^{235}U was determined from the 143.8 keV, 163.3 keV and 205.1 keV lines. ^{238}U was determined from its daughter nuclides ^{234}Th (63.3 keV) and ^{234}Pa (1001.3 keV). ^{228}Th was determined from its short-lived daughter ^{212}Pb, ^{212}Bi and ^{208}Tl.

Results and discussion

The results of activities in sludge from waterworks are given in Table 2. The ratio of ^{137}Cs to ^{90}Sr in sludge in Pitkäkoski was compared to the ratio of ^{137}Cs to ^{90}Sr in surface water in Lake Päijänne. The ratio in the water of Päijänne varied from 180 in 1986 to 5 in 1994. The much higher ratio of ^{137}Cs to ^{90}Sr in sludge at the Pitkäkoski waterwork shows also the much higher accumulation of ^{137}Cs than that of ^{90}Sr into sludge. The results also show the higher mobility of ^{90}Sr than ^{137}Cs in the environment.

Table 2. Concentration of natural radionuclides and ^{90}Sr and ^{137}Cs in sludge at the waterworks.

Waterworks	Treat-ment method	Sample	Date	Bq kg^{-1} dry matter								
				^{228}Ra	^{228}Th	^{226}Ra	^{234}Th	^{234m}Pa	^{235}U	^{90}Sr	^{137}Cs	$^{137}Cs/^{90}Sr$
Helsinki, Pitkäkoski	Al-sulph	accumulating sludge	13.8.1986							81,1	20300	250
(water from Päijänne)	Al-sulph	removed sludge	28.8.1986	0	0		2340	3100	152	95,5	56800	595
	Al-sulph	removed sludge	25.9.1986	0	0		2710	3160	151	61,9	27700	447
	Al-sulph	removed sludge	19.11.1996							40,3	11400	283
	Al-sulph	accumulating sludge	20.11.1991	20	25			2340	146			
	Fe-sulph	sludge after sand filtrati-on	29.5.2001		10		680	1150	59			
	Fe-sulph	removed sludge	29.5.2001		10		510	1000	53	0,91	94,1	103
(water from Vantaanjoki)	Fe-sulph	removed sludge	4.12.2001	27	40		470	780	45	0,71	72,8	103
Espoo, däm-man	Al-sulph		11.11.1991	24	34	140		190				
	Al, Fe	older sludge	30.5.2001	20	26					1,63	63	39
	Fe	sludge	30.5.2001	13	23		48	90	7	0,9	41,8	46
	Fe	sludge	22.10.2001	9	21		39		4			
Tampere, Rusko		removed sludge	21.8.1986							17,7	8700	492
(water from Roine)		accumulating sludge	21.8.1986							42,1	12300	292

In the Dämman sludge , the slight differences were noticed in the concentrations of natural radionuclides in samples taken in 1991 and 2001. In Pitkäkoski sludge the activities were about three times lower in 2001 than in 1991 but about ten times higher than those in Dämman. Perhaps the raw water flowing via a rock tunnel causes a special source of natural activity in the precipitation produced in Pitkäkoski. When water was taken directly from Vantaanjoki river without transportation via a rock tunnel, the concentrations of thorium, protactinium and uranium were slightly decreased. Gäfvert et al.[5] noticed that a general trend for uranium and thorium is that $Fe(OH)_3$ precipitation has a slightly lower removal efficiency than $Al(OH)_3$ precipitation, but this explains only partially the differences in Pitkäkoski in various years. Because the interest was earlier only in artificial radionuclides, the measuring time and the volume of sample was not always enough for the determination of natural radionuclides.

Table 3. Average concentrations of natural radionuclides in sewage sludge in 1991 and 2001.

Wastewater treatment plant	Year	Bq kg^{-1} dry matter						
		^{228}Ra	^{228}Th	^{226}Ra	^{234}Th	^{234m}Pa	^{235}U	^{40}K
Helsinki, Kyläsaari	1991	46	20		430	450	40	210
Helsinki, Viikinmäki	2001	35	17	120	370	460	24	130
Espoo, Suomenoja	1991		20				8	
Espoo, Suomenoja	2001	42	17	84	92	140	7	80
Rauma, Maan-päänniemi	1991	46	26		39		4	140
Pori, Luotsinmäki	1991	51	32		48		6	110
Kotka, Mussalo	1991	78	29		52			150
Kotka, Sunila	1991	87	45		89			110
Loviisa, Vårdö	1991	120	57		99		10	
Loviisa, Vårdö	2001	81	51	82				75

Table 4. Concentration of ^{90}Sr and ^{137}Cs in dewatered sewage sludge.

Wastewater treatment plant		Data	Bq kg^{-1} dry matter		Wastewater treatment plant		Date	Bq kg^{-1} dry matter	
			^{90}Sr	^{137}Cs				^{90}Sr	^{137}Cs
Helsinki	Kyläsaari	22.5.1986	56,7	4940	Kotka	Sunila	8.8.1994	2,63	61
Helsinki	Viikinmäki	22.4.1996	16,0	50	Kotka	Sunila	29.8.1994	3,84	41
Espoo	Suomenoja	22.10.2001	6,48	11,4	Kotka	Mussalo	29.8.1994	1,45	35
Rauma	Maanpään-niemi	2.6.1986	53,6	4460	Loviisa	Vårdö	2.6.1986	15,9	3460
Rauma	Maanpään-niemi	6.6.1986	52,1	3420	Loviisa	Vårdö	14.8.1986	31,1	3390
Rauma	Maanpään-niemi	24.10.1994	9,62	27	Loviisa	Vårdö	12.10.1994	3,75	52,6
Pori	Luotsinmäki	26.5.1986	79,1	5400					
Pori	Luotsinmäki	26.10.1994	7,49	51					

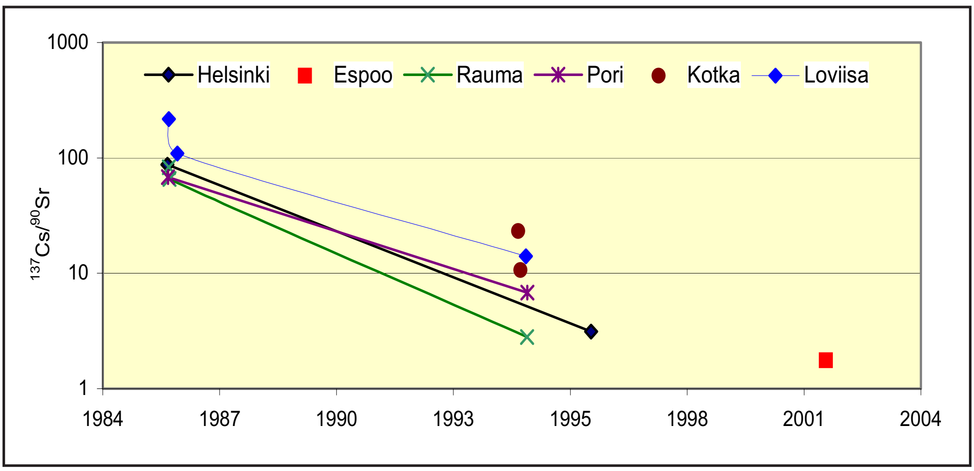

Figure 1. ^{137}Cs/^{90}Sr in dewatered sewage sludge.

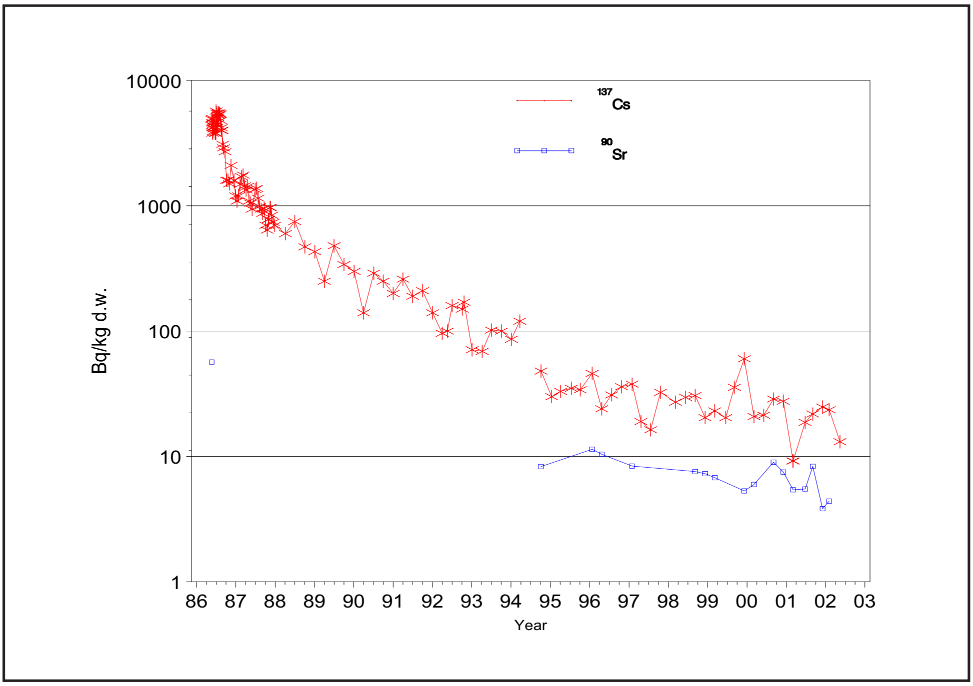

Figure 2. Concentrations of ^{90}Sr and ^{137}Cs in raw sewage sludge at Helsinki Kyläsaari and Viikinmäki.

The activity concentrations of ^{90}Sr in output sewage sludge and ^{137}Cs in the same samples are given in table 4 and figure 1. The concentration of ^{137}Cs and ^{90}Sr in raw sewage sludge in Kyläsaari, Helsinki, (until 1994) and Viikinmäki (since 1994) are shown in Figure 2. ^{90}Sr was analysed only in few samples.

Table 3 summarises the average values of natural radionuclides in sewage sludge taken at various treatment plants in 1991 and 2001.

The activity of tap water and foodstuffs influence the activity concentration of natural radionuclides in sewage sludge as do the precipitation produced in waterworks and transported to wastewater treatment plants and the sludge from industrial processes.

The concentrations of ^{234}Th and ^{234m}Pa, belonging to ^{238}U series, and ^{235}U were higher in the Kyläsaari and Viikinmäki wastewater treatment plants than in Suomenoja, Espoo. Al and Fe precipitation taken from the Pitkäkoski waterworks has increased the amounts of natural radionuclides in sewage sludge at the Kyläsaari and Viikinmäki treatment plants. A considerable part of the new trunk sewers in Helsinki have been constructed as rock tunnels and hence rock water can also infiltrate into sewage water from there.

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3e Natural radioactivity in selected waterworks and private wells in Sweden - a pilot study

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Introduction

Naturally occurring radioactive elements are present in all ground waters. Groundwater from rock aquifers can contain relatively high concentrations of natural radionuclides, especially in areas with uranium-rich granites and pegmatite. Sweden has relatively high levels of natural radioactive elements in the groundwater. In this investigation we have carried out measurements in order to estimate alpha- and beta-emitting natural radionuclides. The alpha-emitting nuclides are primarily ^{238}U, ^{234}U, ^{226}Ra and ^{210}Po. The most important beta-emitting nuclides are ^{210}Pb and ^{228}Ra.

In the European Drinking Water Directive, 98/83/ EC, a reference value of 0.1 millisievert, mSv, per year for Total Indicative Dose, TID, from consumption of drinking water is given. TID includes all radionuclides in drinking water, both artificial and natural, with the exception of radon, radon progeny, ^{40}K and tritium. A first indication of that TID might exceed 0.1 mSv is that the gross alpha activity concentration exceeds 0.1 Bq/l or that the gross beta activity concentration exceeds 1 Bq/l.

A new Swedish Drinking Water Ordinance implementing the new directive has come into force but will not be put into practice until December 2003 (SLV FS 2001:30). Sweden has had action levels for radon in drinking water since 1997 (SLV FS 1997:32). The action level for radon in water from public water supplies is 100 Bq/l. For private wells the recommended maximum level is 1,000 Bq/l.

Sweden has about 2,100 public water supplies and 200,000 private drilled wells used permanently and another 100,000 used in summerhouses. Altogether about 1.2 million of the Swedish population rely on private water supplies.

Objectives and scope

The objective of the investigation has primarily been to test the measurement methods needed to verify compliance with the TID reference value of 0.1 mSv per year, and secondly to get an indication of what kind of levels of TID that consumption of water from waterworks using groundwater could cause in Sweden. Measurements have also been performed of water samples from private wells. Formally, the Drinking Water Ordinance is in most cases not applicable to water from private wells, but it is inevitable that owners of private wells will judge the quality of their water according to the standards of the Drinking Water Ordinance.

Thirty waterworks, chosen at random, were asked whether they were willing to participate in the investigation. 24 of them accepted and delivered a total of 41 water samples. Some of the waterworks took samples from several water catchments and in some cases both from raw water and production water. About 30 water samples from private wells have been analysed, most of them collected in the Stockholm area. Radon was also analysed in these samples.

Sampling and measurement methods

The water samples were collected in clean glass bottles and immediately sent to SSI with information about the waterworks and the samples.

The measurements were performed in a low-background liquid scintillation spectrometer (Wallac 1220 Quantulus) using “Pulse Shape Analysis” to separate the alpha and beta pulses into separate energy spectra, see fig. 1.

Radon measurements:

10 ml of water was slowly poured using a pipette into a low-potassium borosilicate glass vial that was filled with 12 ml of cocktail Ultima Gold XR. The sample was measured after a few hours and the measurement was repeated later to check for leakage of ^{222}Rn from the vials. All alpha pulses are used for the determination of the radon concentration. The results from this method agree well with other methods.

Alpha and beta measurements:

2x19 ml of the water sample was evaporated with a freeze-dryer in a Teflon-coated polyethylene vial and the residue was dissolved in 1 ml of 0,1 M HCL, after which 21 ml of the cocktail Optiphase HiSafe 2 was added. The sample was measured after one month. During that time ^{222}Rn and the short-lived daughters establish equilibrium with ^{226}Ra. Besides the long-lived alpha nuclides ^{238}U, ^{234}U, ^{226}Ra and ^{210}Po, the alpha energy spectrum also contains structure from ^{222}Rn, ^{218}Po and ^{214}Po. As the alpha energy of ^{214}Po is different from all other natural alpha emitters expected in water samples, the ^{214}Po peak (see fig 2) is used to calculate the ^{226}Ra activity and also for calculating the gross alpha activity concentration.

The gross beta activity concentration is calculated using the total beta counts in an energy interval excluding ^{3}H, reduced by the contribution from short-lived radon progeny.

The presence of ^{210}Pb, in significant amounts, can be detected as a peak in the beta spectrum, but the activity determination is associated with large uncertainties.

Assuming that no ^{210}Po is present, the activity of both ^{238}U and ^{234}U can be estimated as the gross alpha activity minus ^{226}Ra activity, but the results can have significant uncertainties due to this assumption. Chemical analyses are performed on a few samples to determine gross alpha, ^{226}Ra, U and ^{210}Pb, and the correlation with these two methods is good.

Results

Waterworks						
		Rn Bq/l	Gross alpha Bq/l	Gross beta Bq/l	Ra-226 Bq/l	U mg/l
Groundwater without artificial infiltration. 15 samples	Median		0,055	0,094	0,009	0,002
	Max		0,94	0,92	0,04	0,037
Groundwater with artificial infiltration.12 samples	Median		0,047	0,14	0,019	0,001
	Max		2,25	1,24	0,34	0,078
Surface water. 5 samples	Median		0,004	0,034	u.d	0,0002
	Max		0,03	0,3	0,006	0,001
Private Drilled wells 21 samples	Median	181	0,71	0,98	0,057	0,023
	Max	3746	8,48	5,99	2,53	0,244

Discussion

The results from this study indicate that a number of Swedish waterworks will have difficulties complying with the reference value for TID. The reference value will also be exceeded in a large number of private drilled wells. The results from this study also show that the measurement method used works well. The investigation will continue during 2002 in co-operation with SGU.

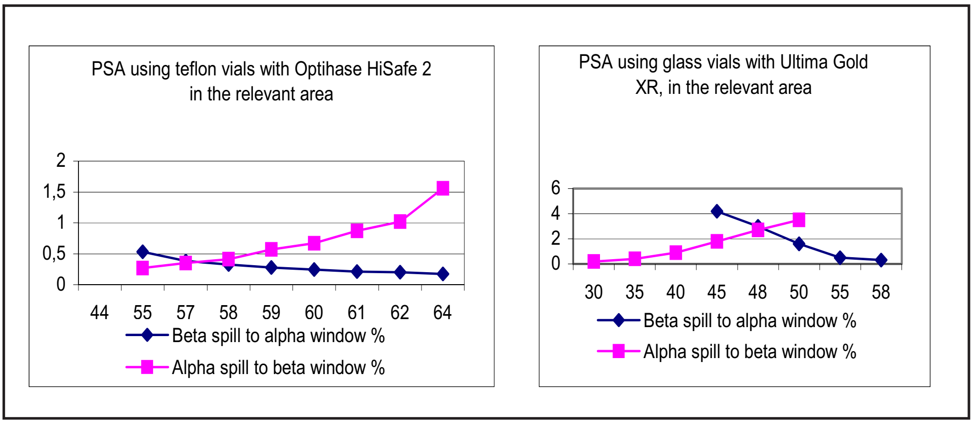

Figure 1.

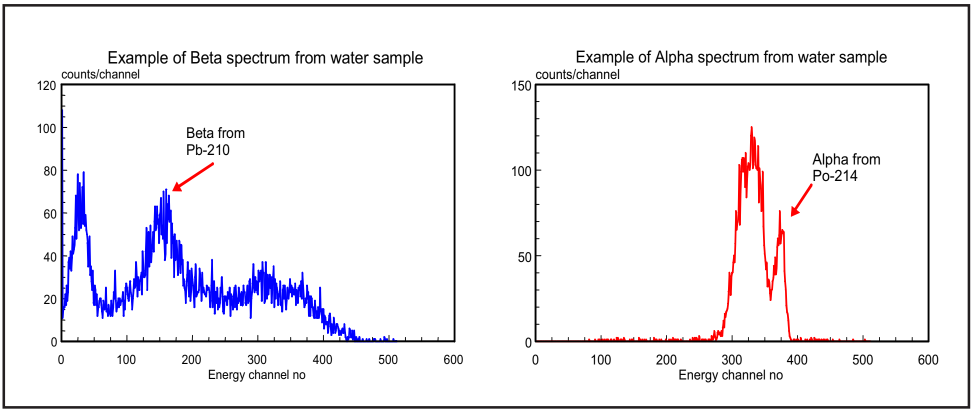

Figure 2.

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3f Seasonal variation of Lead-210 in the air at Ny-Ålesund, Svalbard

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Abstract

In this project we have collected high-volume aerosol samples at Mt. Zeppelin Global Atmosphere Watch station, Ny-Ålesund, Svalbard and measured their lead-210 contents in the laboratory. The measurement is based on the alpha counting of the in-grown polonium-210 using an automatic alpha/beta analyser. The gathered data of atmospheric lead-210 concentrations can be used as a tracer in the analysis of air pollutant concentrations in the area. In the long run the data on atmospheric ^{210}Pb can help to identify natural, e.g. due to the North Atlantic Oscillation (NAO), and anthropogenic variations in the transport behaviour of air masses and thus also air pollutants in the Arctic region.

Introduction

During the past three decades, there has been increasing interest in the presence of airborne pollutants in the Arctic region. However, the interpretation of the results has suffered from the lack of data concerning the atmospheric and coupled oceanic/atmospheric transport processes in the area. In this project we have measured concentration of lead-210 (^{210}Pb) in the air at Svalbard. The gathered data can be used as a tracer in the analysis of air pollutant concentrations in the area. In the long run the data on atmospheric ^{210}Pb can help to identify natural, e.g. due to the North Atlantic Oscillation (NAO), and anthropogenic variations in the transport behaviour of air masses and thus also air pollutants in the Arctic region.

Lead-210 is formed in the atmosphere from the radioactive noble gas radon-222 (^{222}Rn) emanating from the Earth's crust. 99 % of the airborne ^{222}Rn originates from land and only 1 % from the sea (*Baskaran et al*., 1993). Owing to the long half-life (22 years) of ^{210}Pb, its removal from the atmosphere is governed by the different scavenging processes affecting the aerosol particles carrying it rather than radioactive decay. Based on the activity ratio of ^{210}Pb and its progeny, mean aerosol residence times of one to two weeks have been obtained (*Mattsson*, 1975; *Samuelsson et al*., 1986; *Papastefanou and Bondietti*, 1991).

Preiss et al. (1996) have reviewed the publications of ^{210}Pb activity concentrations in the air. High ^{210}Pb concentrations are found in continental air masses. Lead-210 has been used as a tracer for particle-bound sulphate because they are both secondary aerosols, i.e. produced in the atmosphere from their gaseous precursors ^{222}Rn and SO_2, respectively (*Turekian et al.*, 1983; *Mattsson et al.*, 1993).

Recently it was discovered that the several-year oscillation of the ^{210}Pb activity concentrations in the air in Southern Finland is connected to the state of the northeastern part of the Atlantic Ocean. Low ^{210}Pb activity concentrations are associated with the more frequent arrival of maritime air masses in Finland. On the other hand, high ^{210}Pb concentrations are associated with the more frequent presence of continental air masses. Higher amounts of warm and saline water in the North Atlantic Ocean are closely connected to enhanced cyclonic activity and low ^{210}Pb air concentrations in Finland (*Mattsson et al.*, 1996a; *Paatero et al.*, 1998; *Paatero et al*., 2000).

Materials and methods

The sampling site was at Mt. Zeppelin Global Atmosphere Watch (GAW) station, Ny-Ålesund, (78°58´ N, 11°53´ E), on the west coast of Spitsbergen, the largest island in the Svalbard archipelago (*NILU*, 2002; *WMO*, 2002). The station is located 474 m above sea level. It is owned and operated by the Norwegian Polar Institute. The Norwegian Institute for Air Research (NILU) is responsible for the scientific programmes at the station. Meteorological Institute of the Stockholm University co-operates closely with NILU in developing the scientific activities and programmes at the station.

Results and discussion

The observed ^{210}Pb activity concentrations at Mt. Zeppelin, Ny-Ålesund, Svalbard vary between 11 and 620 µBq/m^3 (Fig. 1). A Japanese research group observed ^{210}Pb activity concentrations ranging from 83 to 1204 µBq/m^3 at Ny-

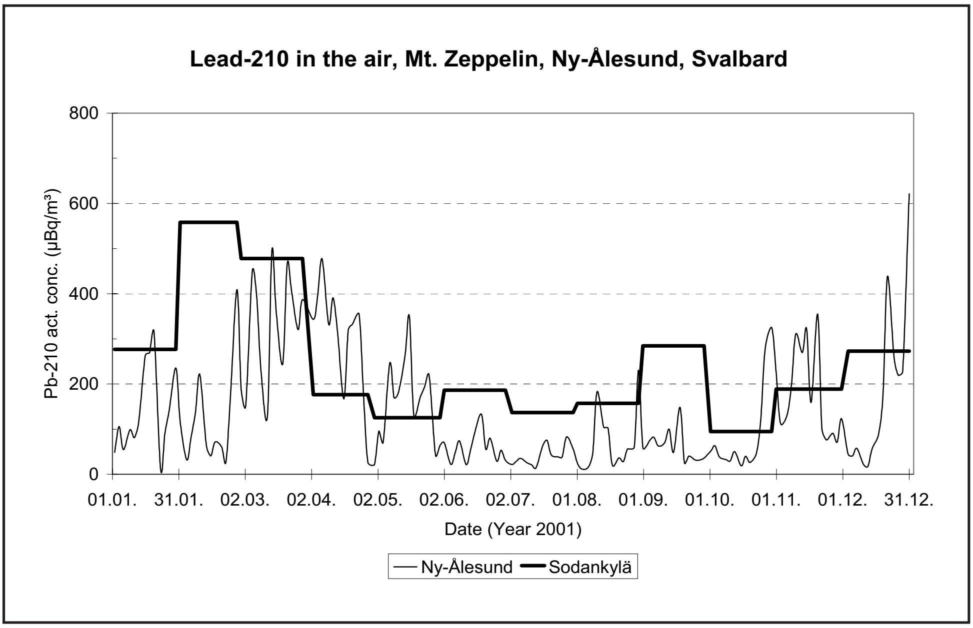

Figure 1. ^{210}Pb activity concentration ($\mu Bq/m^3$) in the air at Mt. Zeppelin GAW station, Ny-Ålesund, Svalbard in 2001.

Ålesund in February-March 1995 (*Suzuki et al.*, 1996). They also reported aerosol residence times ranging from 26 to 78 days based on the $^{210}Po/^{210}Pb$ activity ratio. These values are much higher compared to studies made in USA and Finland, where the residence times have usually been one to two weeks (*Mattsson*, 1975; *Papastefanou and Bondietti*, 1991; *Balkanski et al.*, 1993). *Samuelsson et al.* (1986) reported an average concentration of 75 µBq/m³ in July-September 1980 between the 75th and 83rd latitude north and between Greenland and Frans Josef Land.

The lowest ^{210}Pb concentrations are found in summer and the highest ones in spring. This differs from the seasonal behaviour of ^{210}Pb in Finland, where the maximum concentrations occur usually in winter. The lag between the winter maxima at Mt. Zeppelin and Sodankylä, northern Finland is about a month. Earlier it has been shown that in winter the air masses coming from Arctic regions to northern Finland contain relatively high amounts of ^{210}Pb (*Paatero and Hatakka*, 2000). This was attributed to the small amount of precipitation, reduced air chemistry and stagnant mixing conditions in the troposphere during the Arctic night. These factors increase the aerosol residence time and thus the accumulation of ^{210}Pb into the air. The one month phase shift between Svalbard and northern Finland may be related to the strength of solar radiation and its capability to cause vertical mixing of the air.

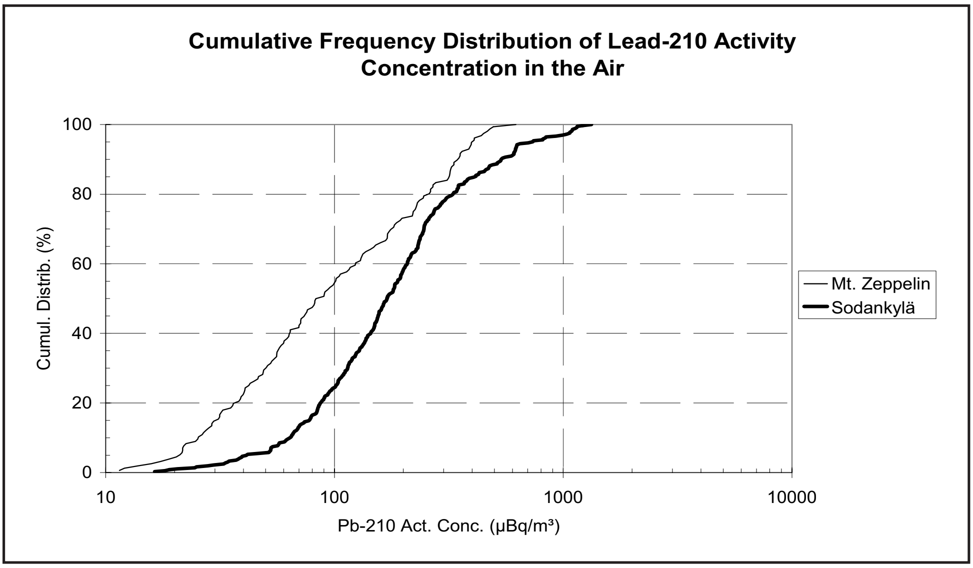

Figure 2. Cumulative frequency distribution of ^{210}Pb activity concentrations at Mt. Zeppelin, Svalbard and Sodankylä, northern Finland in 2001.

The 25, 50, and 75 % percentiles of the ^{210}Pb activity concentrations at Mt. Zeppelin are 42, 83, and 220 µBq/m^3 (Fig. 2). The values are clearly lower than at Sodankylä with corresponding values of 100, 170, and 270 µBq/m^3. The arithmetic mean concentrations in 2001 were 144 and 245 µBq/m^3 at Mt. Zeppelin and Sodankylä, respectively.

The future work will include the comparison of the ^{210}Pb observations to e.g. sulphate and aerosol particle concentrations and the studies on the relation between ^{210}Pb and various meteorological parameters.

Acknowledgements

The authors would like to thank the Norwegian Polar Institute and the Norwegian Institute for Air Reasearch for a pleasant cooperation. The financial support of the European Community – Access to Research Infrastructure action of the Improving Human Potential Programme is gratefully acknowledged.

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SESSION 4. REACTOR SAFETY AND WASTE MANAGEMENT

Radiation protection experience from a fuel failure at Halden reactor

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Abstract

On January 28 2001 lack of cooling resulted in failure of several fuel rods in a test assembly in the Halden reactor. The resulting contamination of the primary circuit lead to very high radiation levels in the containment of the reactor. There were no releases of radionuclides to the containment or to the environment around the reactor plant. In the subsequent handling of the failed fuel and bringing the reactor back to a normal state, the primary aim was to keep radiation doses to the workers and releases to the environment as low as possible. Every operation was analysed and carried out according to these criteria. Consequently the collective dose from the whole operation was 0.052 manSv, which is 10% of the average annual collective dose at the Halden reactor, and there were no additional releases of radionuclides to the environment.

Introduction

In the evening Sunday January 28th 2001 lack of cooling resulted in a simultaneous failure of several fuel rods of a test assembly in the Halden reactor. This activated one of the reactor's safety systems with subsequent control rod injection and closing down of the reactor. At the same time a substantial increase the gamma dose rate was seen on the monitors in the reactor containment, with several going off scale. The released activity from the failed fuel rods was contained in the closed primary circuit of the reactor, and no activity was released to the containment or to the reactor surroundings. The increased dose rate level in the containment was due to external radiation from primary circuit components.

The following cleaning up and preparing the reactor for further operation took more than three months. It was very important that this process could be accomplished with as low personnel radiation doses as possible, otherwise it could influence future possibilities of maintenance and replacement of experiments. On the other hand was the contradictory goal of getting the reactor operational again as quickly as possible. The first objective was therefore to reduce the dose rate from external radiation in the containment. In addition a meticulous control of the personnel radiation doses through out the whole operation was introduced.

External dose rate development

Figure 1 shows the reactor containment with reactor tank and steam transformers. Most of the work in cleaning up after the incident and unloading the damaged assembly would be performed from the rotating lid above the reactor tank.

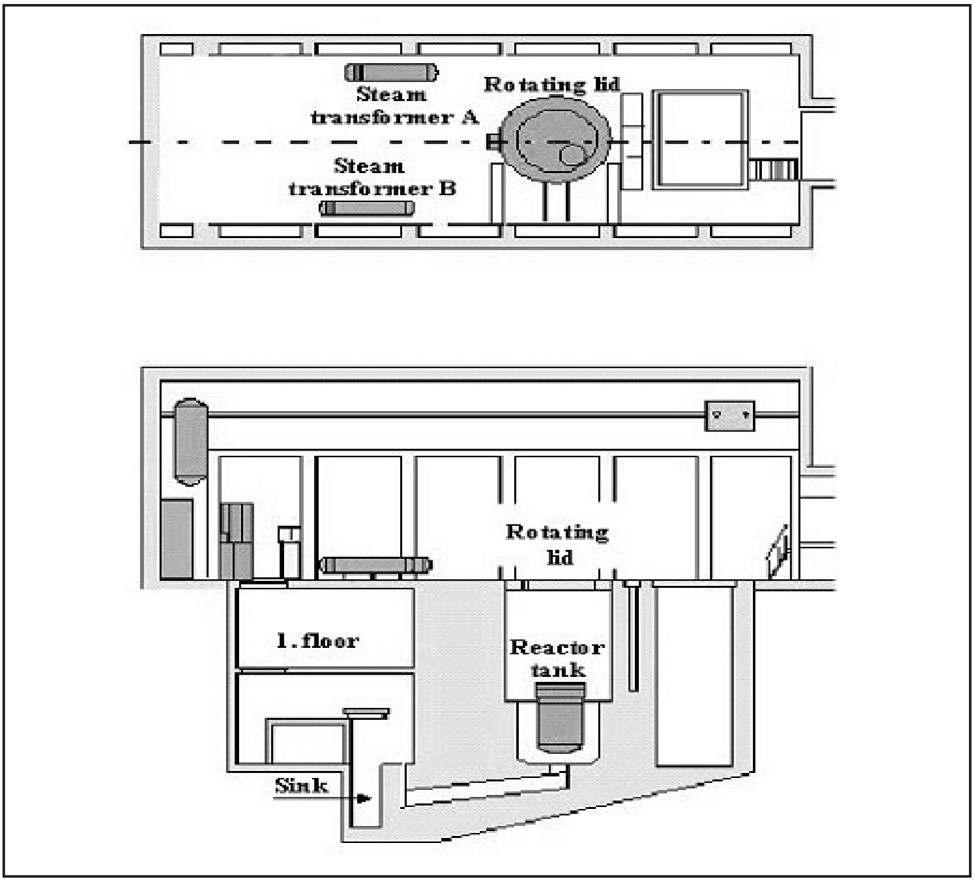

Figure 1. Reactor containment.

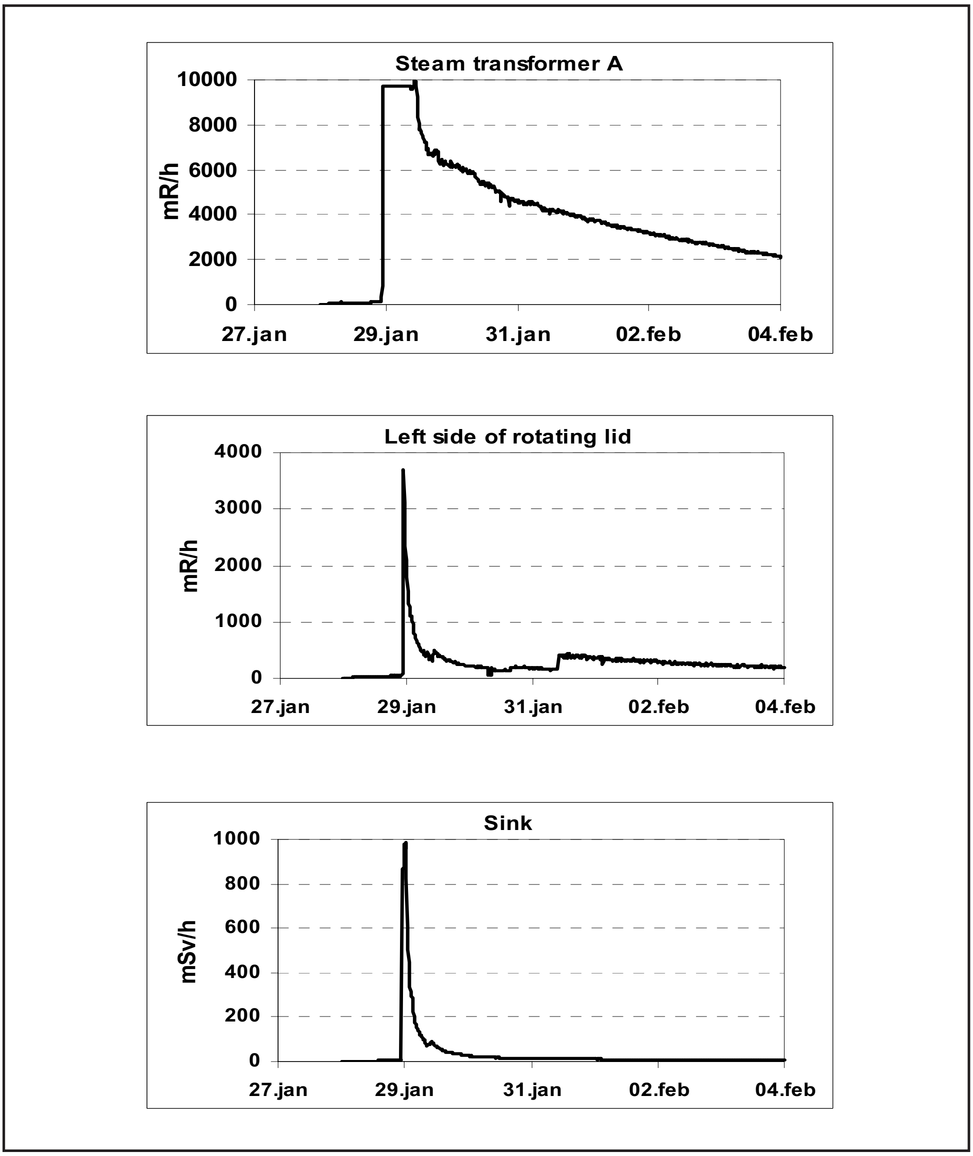

Figure 2. Dose rates at different locations in the containment

The gamma monitors showed an initial very rapid decrease, which levelled off after two days, as can be seen on Figure 2. Two days after the incident measurement of dose rate in the working area was carried out. The dose rates were then 30 mSv/h between the steam transformers and about 3 mSv/h on the rotating lid. The main source of the radiation in the working area was the steam transformers. The decrease of the radiation from the steam transformers showed that the dominant nuclides had an effective half-life of 4 to 5 days. Measurement of the lead half value layer of the radiation indicated an effective energy of approximately 350 keV. A combination of 90% ^{133}Xe and 5% each of ^{132}Te with daughter ^{132}I, and ^{131}I would give this half-life and effective energy.

The dose rate in the working area was far too high to initiate any work, and it was decided to shield the rotating lid from the steam transformers. Two rows of 2 meter high and 5 cm thick steel plates were positioned between the rotating lid and the steam transformers, giving a dose rate reduction on the rotating lid of about 40%.

Fourteen days after the incident, shielding and disintegration had brought the dose rate in the working area down to 0.05 to 0.1 mSv/h. The primary circuit purification system had been fully operational throughout the period, and the activity concentration in the primary circuit water was brought down to values comparable to values experienced in shut down periods before the incident. It was therefore considered justifiable to start the clean up operation.

The clean up operation and unloading of the damaged assembly

Visual inspection with a periscope in the reactor tank showed that fuel fragments were scattered out on the reactor tank bottom plate, both in the form of particles, but also as a thin covering. There was some uncertainty as to whether the covering was actually fuel, but gamma spectral analysis of a sample of the covering clearly showed a radionuclide content as would be expected from the fuel from the failed rods. The covering was later identified as UO_2CO_3, and it was expected that it would be dissolved at temperatures above 150 ºC. The clean up process therefore consisted of a mechanical removal of the fuel particles followed by electrical, and later nuclear, heating of the primary circuit water for dissolving the covering. The dissolved UO_2CO_3 would then be removed from the water in the reactor's purification circuit.

Removal of fuel particles from the reactor tank bottom plate and unloading of the damaged assembly was carried out over the next two months. Thereafter the heating process described above was performed for fourteen days.

The Norwegian Radiation Protection Authority (NRPA) had been continuously informed of the situation and the clean up process. Based on a comprehensive analysis of all possible consequences regarding nuclear safety and expected releases of radionuclides to the environment with further reactor operation, NRPA gave consent to further operation three months after the incident.

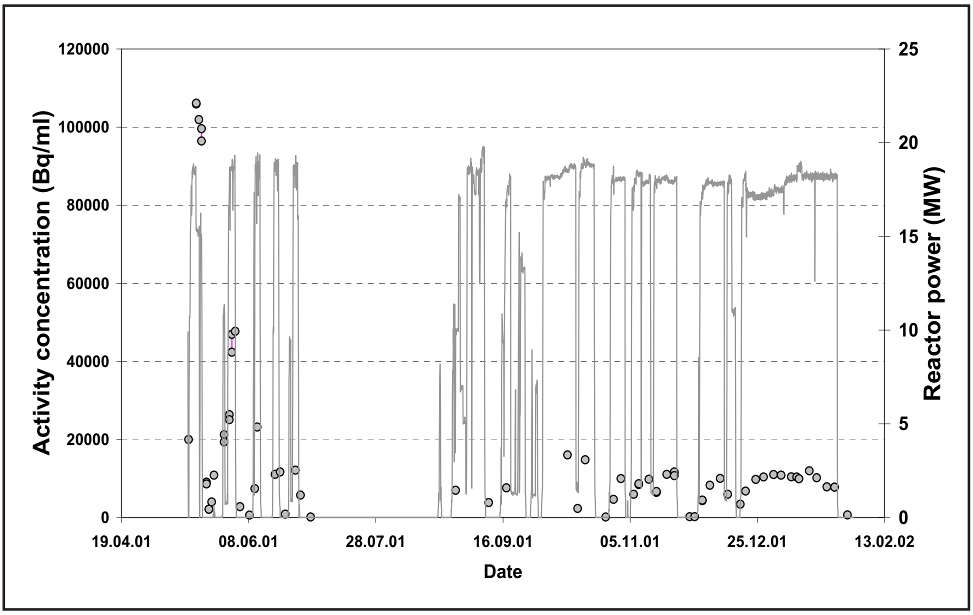

Figure 3. Activity concentration of ^{239}Np (markers) and reactor power (curve) from May 2001 until February 2002.

The collective radiation dose to the personnel involved in the clean up operation amounted to 0.052 manSv. This is approximately 10% of the average annual collective dose at the Halden reactor. The operation did not cause any additional releases of radionuclides to the environment.

Further development of radionuclide concentration in the primary circuit water

Figure 3 shows the concentration of ^{239}Np in the primary circuit water from start of operation until shut down in January 2002. The concentrations of fission products were very high in the first periods of reactor operation, but decreased rapidly to about 15% of the initial concentrations. The high initial concentrations were due to the dissolving process of the remaining UO_2CO_3, which during the process was brought into the high neutron flux in the reactor core. Further decrease in radionuclide concentrations is slow, and the concentrations are still about four times higher than before the incident. The source of the present activity is a general uranium contamination of the reactor core, which will be reduced with further reactor operation and future unloading of contaminated fuel assemblies. The latter can be seen in Figure 4, where the

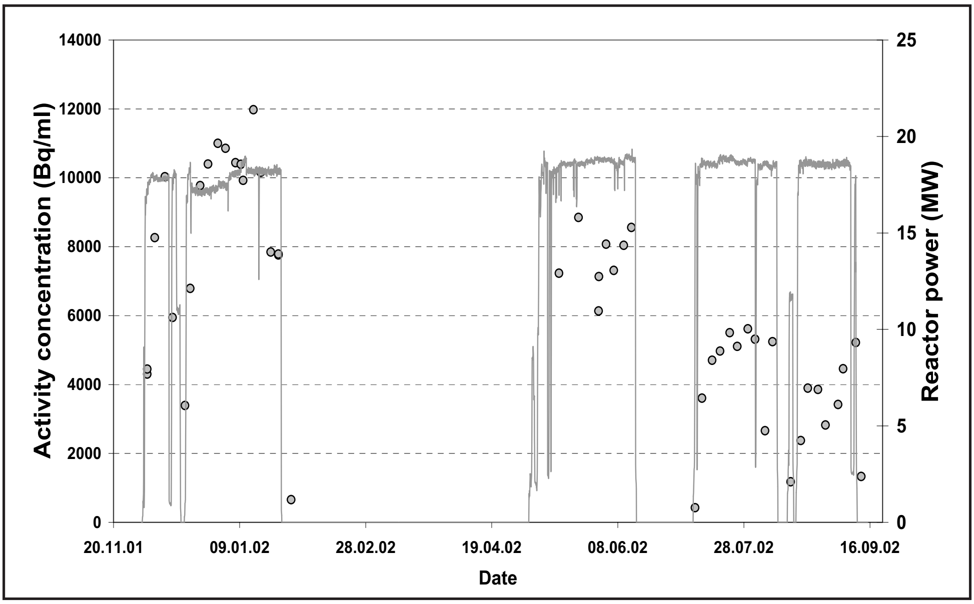

Figure 4. Activity concentration of ^{239}Np (markers) and reactor power (curve) from May 2001 until February 2002.

effect of unloading several contaminated fuel assemblies in the June 2002 shutdown can be seen as a general reduction of the activity concentration in the following reactor operation.

Dekontaminering av eksperimentalkrets ved Haldenreaktoren

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Sammendrag

Ved Haldenreaktoren er en lukket krets for testkjøring av brensel og materialprøver blitt dekontaminert. Kretsen ble dekontaminert ved vekselvis behandling med $KMnO_4$ og oksalsyre etterfulgt av skylling, i flere omganger. Eksperimentet var meget vellykket. Under forsøket oppstod det ingen lekkasjer eller andre problemer og i ettertid er det ikke registrert svakheter ved noen av kretsens komponenter som følge av kjemikaliene som ble benyttet. Det gjennomsnittlige strålingsnivået fra kretsen sank med 89 %, og rørbiter som ble tatt ut hadde rene blanke metalloverflater. Fra start til slutt krevde forsøket 16 person milli Sv. Utslipp som følge av forsøket utgjorde kun 1/2000 del av utslippstillatelsen på 1 mikro Sv til vann og 7 ‰ av totalt utsluppet radioaktivitet til vann fra april til desember i 2000[1].

Innledning

Haldenreaktoren er en forskningsreaktor for testing av brensel og materialer. En del av eksperimentene utføres i egne lukkede kretser, som gjør det mulig å utføre forsøkene ved andre reaktorforhold enn reaktorens eget. De lukkede kretsene er bygget opp av en eller flere såkalte flasker, og en ytre del bestående av rørsystemer og komponenter som pumper, heatere/kjølere, ventiler, filtre, ionebyttere etc.. I motsetning til flaskene som befinner seg i selve reaktorkjelen, er den ytre delen plassert i reaktorhallen. Under kjøring av reaktoren sirkulerer det vann gjennom hele kretsen. Vannet tar med seg radioaktive spesier, som i sin tur fester seg til oksidbelegget inne i kretsene. Spesielt komponenter med temperaturgradienter, som heatere/kjølere, men også pumper og ventiler er utsatt for slik kontaminering. Resultatet over tid er høy ekstern stråling i de områdene hvor slike lukkede

[1] IFE fikk ny utsippstillatelse i 2000 som trådte i kraft fra 1. april. Grensen var fortsatt 1 ìSv/år, men modellen som ligger til grunn beregningene var endret.

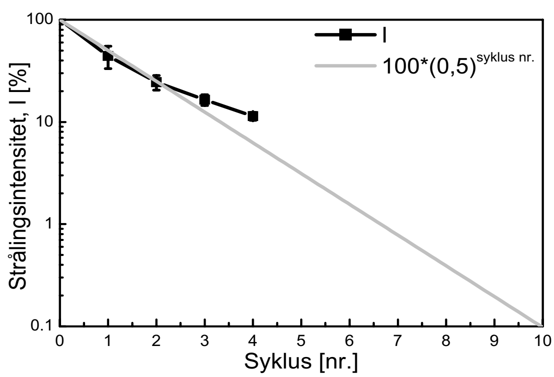
kretser er plassert. For Haldenreaktoren er ekstern stråling fra slike lukkede kretser en av hovedkildene for doser til arbeiderne. Målet med dekontamineringsforsøket var å komme frem til en metode for generell dekontaminering av eksperimentalkretser slik at doser til personell ville reduseres i fremtiden: Metoden måtte ikke skade komponentene i kretsene.

Forsøket

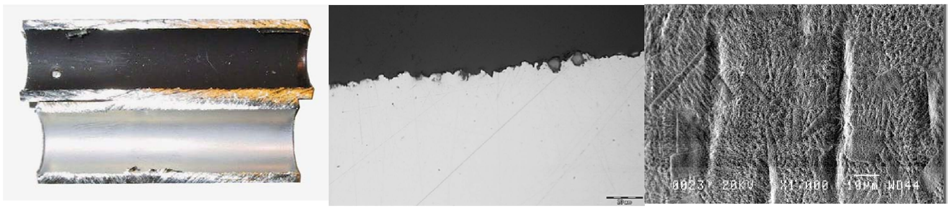
Kretsen som ble dekontaminert, er en eksperimentalkrets som har vært i drift i seks år. I løpet av denne tiden er det utført forsøk ved flere forskjellige reaktorforhold, inkludert forsøk som har gått ut på å få fysisk kontakt mellom brensel og kjølevann. Kretsens rørsystem bestod av syrefast stål. Dekontamineringen ble utført ved å vekselvis behandle kretsen med et sterkt oksidasjonsmiddel ($KMnO_4$) og en svak organisk syre (oksalsyre), etterfulgt av skylling med vann. Disse stegene ble repetert flere ganger (sykluser) inntil ønsket resultat var oppnådd: Å fjerne mest mulig oksidbelegg uten å tære på selve metallet i rørene, da dette ville forringe materialets integritet.

Forsøket forløp uten komplikasjoner. Det oppstod ingen lekkasjer eller andre problemer som følge av slitasje på pumpe eller andre komponenter, og inspeksjon i ettertid har òg vist at alle kretsens komponenter har tålt påkjenningen fra kjemikaliene.

Reduksjonen i strålingsnivå var stor allerede etter første syklus. I snitt gikk det midlere strålingsnivået ned med 56 % (Figur 1). Reduksjonen flatet imidlertid noe ut utover i forsøket, og etter fire sykluser endte det med en senkning i midlere strålingsnivå på 89 %.



Figur 1. Reduksjon i midlere strålingsnivå fra kretsen som følge av dekontaminering.



Figur 2. Bildet lengst til venstre viser rørprøver tatt fra samme sted i kretsen, før og en etter dekontaminering. I midten er et LOM-bilde av tverrsnittet av innvendig overflate av det dekontaminerte røret, og bildet lengst til høyre er et SEM-bilde av innvendig overflate på det samme røret.

Rørbiter ble skåret ut av kretsen både før, under og etter dekontaminering. Visuell in-speksjon viste at rør med mindre dimensjoner (18 mm diameter) raskt fikk blanke og rene overflater, mens i mer grove rør (48 mm diameter) var opprensingen dårligere. Dette var imidlertid eldre rør hvor en òg ville forvente tykkere oksidlag. I tillegg til den visuelle inspeksjonen ble det tatt bilder med scanning elektron mikroskop (SEM) og lysmikroskop (LOM) (Figur 2).

Forsøket krevde arbeid på kretsen både før, under og etter dekontamineringen. Spesielt ombyggingen for å sette kretsen i stand for dekontaminering var arbeids- og dose-krevende. I alt gikk det med 218 persontimer, og 15 person m Sv til denne jobben. Dosene under dekontaminering kom fra drift av kretsen. Dette var lave doser, 0,4 m person Sv, da kretsen i hovedsak opereres utenfra. Arbeidet med å sette kretsen tilbake til opprinnelig stand var for så vidt også tidkrevende, men strålingsnivået i arbeidsområdet var nå betydelig redusert. Arbeid etter dekontaminering ga derfor kun 0,5 person mSv. Totalt gikk det med 16 person mSv på hele forsøket, hvilket utgjorde mindre enn 4 % av kollektiv dosen ved reaktoren i 2000.

Rundt 250 liter vann gikk med i hvert steg. I tillegg til kjemikalier og oppløst oksid, innholdt vannet aktivitet, og måtte renses før det kunne slippes ut. Til dette var det bygget et eget rensesystem bestående av en oppsamlingstank, pumpe, filtre og en kationbytter. Vannet ble sirkulert i dette systemet inntil aktivitet og kjemikalieinnhold var så lavt at det kunne slippes til omgivelsene. Erfaringene her var at opprensingen var god, men den fysiske ut-formingen førte til at det tok lang tid. Utslippene som følge av dekontaminering av kretsen, ble lave i forhold til det som slippes ut på årsbasis. Den teoretiske stråledosen til individer i utsatt gruppe ble $5\cdot10^{-4}$ µSv, hvilket kun utgjorde 1/2000 del av utslippstillatelsen til vann på 1 µSv til

Tabell 1. Teoretisk dose til individer i utsatt gruppe som følge av dekontaminering [μSv]

Nuklide	fra dekontaminering	totalt 1/4-31/12 2000
^{134}Cs	$4{,}0 \cdot 10^{-6}$	$2{,}2 \cdot 10^{-4}$
^{137}Cs	$2{,}4 \cdot 10^{-5}$	$2{,}9 \cdot 10^{-3}$
^{60}Co	$2{,}6 \cdot 10^{-4}$	$5{,}9 \cdot 10^{-2}$
^{51}Cr	$6{,}8 \cdot 10^{-7}$	$1{,}7 \cdot 10^{-4}$
^{106}Ru	$4{,}6 \cdot 10^{-6}$	$8{,}1 \cdot 10^{-6}$
^{125}Sb	$1{,}8 \cdot 10^{-4}$	$6{,}3 \cdot 10^{-4}$
^{144}Ce	$3{,}0 \cdot 10^{-6}$	$3{,}0 \cdot 10^{-4}$
^{95}Zr	$1{,}1 \cdot 10^{-5}$	$2{,}6 \cdot 10^{-3}$
^{95}Nb	$1{,}2 \cdot 10^{-5}$	$1{,}8 \cdot 10^{-3}$
Totalt	$\mathbf{5 \cdot 10^{-4}}$	$\mathbf{7 \cdot 10^{-2}}$

utsatt gruppe og 7 ‰ av utsluppet aktivitet fra april til desember i 2000. For nuklidene ^{106}Ru og ^{125}Sb, stod derimot dekontamineringen for store deler av utslippet av disse nuklidene det året (Tabell 1). Nuklidene var først mulig å se i det rensede vannet da mengden ^{60}Co i kretsen overskygget tilnærmet all annen aktivitet. Opprensingen av Ru og Sb gjennom ione-bytteren må derfor ha vært vesentlig dårligere enn for Co. En erfaring som ved nye dekontamineringer har ført til endringer i rensesystemet.

Diskusjon og konklusjon

Selve dekontamineringen av kretsen var meget vellykket. Strålingsnivået ble betydelig redusert, og kom ned på nivå med resten av reaktorhallen. Kretsens komponenter tok ikke skade og det oppstod ingen komplikasjoner underveis. I tillegg til de gode resultatene for reduksjon i stråling, viste forsøket at det er mulig å gjennomføre en dekontaminering av en eksperimentalkrets uten økning i utslipp av betydning. Forsøket krevde en del dose, men totalt sett ble kollektivdosen lavere enn forventet. Dessuten er 80 % av denne dosen påbeløpt under arbeid med å bygge rensesystemet, et system som skal benyttes i alle dekontamineringer fremover, og derfor en engangs investert dose. Beregninger viser også at antallet person Sv påbeløpt vil være spart inn i løpet av 264 timers arbeid i det samme området av reaktorhallen som konsekvens av strålingsreduksjonen dekontamineringen gav. Til sammenligning er en langstopp gjerne på 2000 til 3000 persontimer.

Alt i alt er derfor inntrykket positivt. Det som gjenstår å se, er hvor fort ny aktivitet fester seg til de dekontaminerte rørveggene. I ettertid er imidlertid eksperimentalkretsen revet, men nok en eksperimentalkrets er siden dekontaminert. Denne kretsen skal inn med nye eksperimenter ved slutten av 2002, og det ses frem til de erfaringer som dette vil gi.

Særskilt takk til:
Jan Arvesen som har stått for den praktiske kunskapen om selve dekontamineringen, og Marit Espeland, IFE Kjeller, som har tatt SEM- og LOM-bilder.

The Biosphere Research at SKB

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Abstract

Every third year the Swedish Nuclear Fuel and Waste Management Company (SKB) summarises achieved results and updates the research program for proper handling of radioactive waste in Sweden. In this paper a short summary of the biosphere program is made. Especially a new systems ecological modelling approach to describe dispersal and uptake of radionuclides is described. Moreover an outline of the siting programme is presented.

Introduction

Every third year the Swedish Nuclear Fuel and Waste Management Company (SKB) summarises achieved results and updates the research program for proper handling of radioactive waste in Sweden (SKB, 2001b). In this paper a short summary of the biosphere program is made. For a more detailed program see further SKB (2001b) as well as cited reports. Most of them are available on internet (www.skb.se).

Currently SKB is in an intensive phase to perform field studies at two proposed sites for a potential deep repository of spent fuel. This program includes an environmental impact assessment as well as performance analysis of suggested repository and an encapsulation plant. It will be handed over to the authorities around 2008 according to current time-plans.

Thus in the coming years the biosphere program will handle issues from data collection, data generalisation to modelling and dose estimation. Moreover, the authorities' regulations require that future safety assessments provide a more realistic description of the biosphere and an estimation of the consequences for surrounding fauna and flora (SSI, 1998). Finally the site investigations make the biosphere concrete, which entails that simplifications in how the biosphere is conceptualised must be valid for the site in question.

The following gives an overview how this will be achieved through modelling and some examples of special studies that needs to be performed.

Model development

SKB's modelling of radionuclide transport in the biosphere in the safety assessment have been carried out with the tools BIOPATH and PRISM. Studsvik EcoSafe has developed these with support from SKB since the 1970s. The tools have been utilised for various safety assessments, such as the KBS studies, SFR, SKB 91, SR 97 and SAFE, and have been progressively refined by SKB's efforts, summarised in a review by Edlund et al. (1999) and in the most recent safety assessments (Bergström et al., 1999; Karlsson et al., 2001).

In the safety assessment of a deep repository for spent fuel, SR97, models for typical ecosystems (lake, running water, archipelago, coast, mire, agricultural land and well) were analysed (Bergström et al., 1999). The biosphere for three hypothetical repository sites was divided into squares sized 250 x 250 metres which were associated with one of the typical ecosystems based on available map information (Nordlinder et al., 1999).

The advantage of this approach is that the causes of the variation in estimated dose can be separated. The uncertainty in the data used for the models accounted for a variation of roughly one order of magnitude in dose, while the choice of the typical ecosystem accounted for variations of up to seven orders of magnitude. Uncertainty analyses and sensitivity analyses also showed that the biological processes need to be described better and that the physical parameters need to be measured better (Karlsson et al., 2000).

In the SAFE project, the safety assessment for SFR, the models were further refined for different types of ecosystems and a time-dependent evolution was introduced (Karlsson et al., 2001). In addition, site-specific biosphere data were gathered and investigations were conducted to obtain a better body of data (Kautsky, 2001).

The concepts used in the models above are to a large extent based on the use of generic transfer factors to different compartments, which presumes that the system being modelled is in equilibrium. Furthermore, the transfer factors are based in many cases on empirical data without a mechanical explanation. The models describe the pathways that affect man and human food, while other parts of the biosphere are seldom dealt with.

These simplifications may be warranted for safety assessments where doses to man are overestimated. But they are insufficient for a proper understanding and an explanation of the simplifications. Alternative models are needed to validate the assumptions that are made. Other models are also needed to be able to make use of site-specific information on processes and states in the ecosystems. To estimate the consequences for surrounding fauna and flora in accordance with the regulations, models are needed that are based on the flow of radionuclides in the entire ecosystem and not just for specific pathways that are critical for man, such as well or cow's milk.

The use of process-based models is an appropriate way to solve some of these problems. The transfer between compartments is in these models is based on natural processes such as photosynthesis, degradation, food ingestion, metabolism, nutrient needs, etc. These processes are coupled and the flows are driven for the most part by the mass balance between the fixation and degradation of organic material, which is sustained by other flows of organic and inorganic materials (e.g. oxygen, carbon dioxide, water, and nutrients). Proportional flows of radioactive substances are associated with these flows. The models are general and can be used for all radionuclides. Even if data are lacking for transfer factors, good estimates can be made of the concentration in different compartments and organisms. Another advantage is that the models are scalable to different site and climatic conditions. Many of the conditions are measurable in the field and non-nuclide-specific, e.g. geometry of the drainage basin, water balance, and ecosystem.

For SAFE, new models for the coastal area based on these principles were developed (Kumblad, 1999; 2001). These models are mechanistic and dynamic mass-flux models and were developed for the ecosystem on the hierarchical level of population and uses site data. One of the models is a carbon and nutrient flow model (CNP-model) for the area above the SFR-repository (Kumblad, 2001, Kumblad and Kautsky, in press, Kumblad et al., in press). This model was developed to describe and get an overview of the ecosystem at the site and to identify the most important processes for mass-fluxes of matter. The CNP-model describes the flow of carbon and nutrients between both abiotic and biotic components of the ecosystem. In the model, carbon enters the food web via primary producing organisms and the trophic interactions of the components in the system determine how the matter is channelled through the ecosystem. The processes involved in the CNP model are primary production, respiration, consumption and water exchange.

Because C-14 is one of the most important radionuclides in SFR as a potential dose contributor to the local environment, the CNP-model was extended to describe the fate of a hypothetical leakage of C-14 to the area (Kumblad, 2001, Kumblad et al., in press). The major uptake pathway was through primary producing organisms in proportion to the rate of primary production and the amount of C-14 compared to DIC (dissolved inorganic carbon) in the water. The trophic transfer of the radionuclide was modelled to be equivalent to the rate of consumption for the various organism groups and the C-14 concentrations in their food items. Thus a model without using transfer factors.

Since a multitude of isotopes often need to be considered in safety assessments and sufficient data seldom exists to run transfer factor models, the developed C-14 model was further extended to a generic radionuclide model for the aquatic ecosystem. Systematic analysis of the importance of radionuclide specific mechanisms showed that the adsorption process was of subordinate importance compared to uptake of radionuclide via consumption. Analysis of the excretion efficiency of radionuclides indicates that the magnitude of biomagnification of radionuclides in fish not exceed a factor 3. Calculated bioconcentration factors (BCF) showed fairly good agreement with empirically obtained BCFs for about 40 radionuclides (Næslund et al. in manus).

The process based modelling studies shows that it is possible to apply mechanistic ecosystem models for the fate of radionuclides in the environment and to estimate concentrations in components of the whole ecosystem with only a few radionuclide specific parameters such as plant uptake and adsorption efficiency. The developed model is a first step towards a new type of methodology to be used in assessments of radionuclides in the environment.

Other areas of special concern

In most safety assessments of radioactive waste for the planned repositories the releases of radionuclides (if any) are at low concentrations. That means that only mechanisms accumulating radionuclides for a long time period can create concentrations that could be hazardous for humans and the environment. Such potential high accumulating environments are the forests with its soil profile, mires and sediments in lakes and the sea.

The forest is the dominant ecosystem for the hypothetical sitings (Lindborg and Schüldt, 1998). It is likely that at least the wet forest will be a recipient. The forest has been the focus of several projects that have studied the fallout from Chernobyl (Avila et al., 1998; SSI, 1999). But most studies have mainly been concerned with the short-term consequences of radionuclide transport. Few calculations have been done of the migration and accumulation of radionuclides from a deep repository in forest. The most important long-term processes are accumulation of nuclides in the soil profile and biological leaching processes that move nuclides to biota. The upward transport of radionuclides from the groundwater table into the roots and vegetation is also essential. This needs further studies.

Mires and wetlands are important and probably the most likely recipients for radionuclides, especially in the future. These ecosystems are a probable discharge point from the geosphere and a result of the natural future evolution of the biosphere after post-glacial land uplift in a coastal area. The ontogeny (evolution) of a marine area to a lake and then a mire has been described in connection with the SAFE project (Brunberg and Blomqvist, 2000). Currently a review of important processes in mires is performed, which will lead to a further development of understanding and modelling (Kellner, 2002 in manus).

The sediments in seas, rivers and lakes comprise important areas that influence radio-nuclide transport to biota. In many potential discharge areas, the radionuclides will pass a sediment layer. The permeability and adsorption of the sediment affect the pattern of dispersal and dilution. A marked change in redox conditions, salinity and biological activity takes place in the boundary layer between sediment and water, which can greatly influence the radionuclide flow. In the short term, these processes will probably reduce radionuclide efflux and result in lower doses. In the long term, however, large quantities of radionuclides can accumulate, only to be released later due to land uplift and resuspension, resulting in higher doses. Furthermore, the organisms that live in sediments are exposed to elevated levels, which can then be passed on in the food chains, for example via fish to man.

In connection with the SAFE project, the sedimentation environment in northern Uppland has been modelled from approximately 10,000 years ago to 5,000 years in the future (Brydsten, 1999). The controlling parameters were wave fetch, which has determined the force of wave erosion, and the process of land uplift, which has determined the depth and creation of protective islands and archipelagos.

Also efforts have been made to compile current radioecological knowledge. A literature study reviews how different iodine isotopes occur and how they move in soil (Johansson, 2000). The biosphere parameters for the radio-nuclides in the most recent safety assessments is compiled in a radionuclide catalogue (Karlsson and Bergström, 2002).

Site investigation programme

Inadequate availability and quality of data cause some of the uncertainties in the biosphere. To support the development of models and furnish site-specific data for the safety assessments, data need to be gathered during site investigations.

The site investigation programme for siting of a spent fuel repository started in 2002 will be one of the most extensive data collections ever conducted in Sweden. An overview of the scope of the programme is provided in the background material for SKB, 2000, 2001a. In order to satisfy the need for data and understanding for the safety assessments and the biosphere models, data must be collected from the surface ecosystems. Variables and parameters judged to be important are given in (Lindborg and Kautsky, 2000). Efforts have been co-ordinated with the geosphere programme to find common needs of data for boundary conditions, input data for the environmental impact assessment and background material for future monitoring programmes. Moreover, variables have been identified which are of value for the planning of the site investigation programme in order, for example, to reduce environment disturbances. Various compilations have since been made of existing data and methods within various fields, e.g. meteorology, oceanography, hydrology (Lindell et al., 1999), existing information on agriculture, forestry, population etc. (Haldorson, 2000), methods for studying lakes and rivers (Blomqvist et al., 2000, 2001) and other biological methods (Kyläkorpi et al., 2000). Thus siting programme in combination with other research effort will provide new data, new understanding and new model in the biosphere, but they will create new questions that need to be answered.

Conclusions

SKB is in an intensive phase to improve the understanding of the biosphere during the coming 6 to 10 years. Parallel efforts will be made in developing models, modelling tools, understanding of important processes as well as collecting a vast amount of field data. SKB invites to a participation in reviewing, commenting and discussing the results obtained.

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Retrieval of a near-surface repository at IFE-Kjeller

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Abstract
In 1970 a total of 997 drums and 19 other units containing radioactive waste were buried in clay at IFE-Kjeller. When the Norwegian Parliament in 1994 decided to build a combined storage and repository facility for low- and intermediate level radioactive waste in Himdalen (KLDRA), it was also stated that this waste should be retrieved and transferred to this new repository.

Retrieval of the buried waste was carried out during the autumn of 2001. A short description of the work is given below.

Introduction
In 1970 IFE got permission from the National Institute for Radiation Hygiene (now Norwegian Radiation Protection Authority) to bury approximately 1000 drums of radioactive waste and 19 other waste items in a near surface repository at IFEs premises at Kjeller. When the Norwegian government in 1994 decided to build a new combined storage and repository in Himdalen approximately 30 km from IFE (KLDRA-Himdalen) it was also decided that the waste in the old near surface repository should be retrieved, that the waste drums containing plutonium should be placed in the storage part and that the rest of the waste should be deposited in the new facility.

The work of retrieving radioactive waste from the old near surface repository at IFE started 14 August 2001. The last drum was removed 22 October 2001. Free classification of the area was finished in early November 2002.

Description of the old near surface repository at IFE, Kjeller
A total of 997 drums of radioactive waste and 19 other waste items were buried in 1970. The drums were standard 210 l steel drums with variable shielding and inner drums dependent on the type and activity of the waste.

Complete information of the radioactivity of various radionuclides in the repository was difficult to obtain because of a rather poor documentation of the

Table 1. Radioactivity in the repository by 31 December 1970 and 1 August 2001.

		Radioactivity (GBq)	
Nuclide	Half-life (a)	31 December 1970	1 August 2001
Co-60	5.3	1870	30
Sr-90	28	2580	1180
Cs-137	30	2530	1220
Pu-239	$2.41 \cdot 10^4$	79.1	79.1
Pu-240	6569	5.2	5.2
U (nat.)	$4.47 \cdot 10^9$	1.3	1.3
Other nuclides		2240	< 100
Total		~ 9300	~ 2 600

content in the separate drums. A summary of the total activity of the most important radionuclides is given in Table 1. The total activity of nuclides with half-life > 1 year buried in 1970 was approximately 9300 GBq. In August 2001 prior to the retrieval the total activity had decreased to approximately 2600 GBq.

The waste drums were embedded in clay and stacked in two layers lying sideways [1]. The area covered 11.5 m x 23 m. The upper layer of drums was covered by 1.5 –2 m of clay and soil. In the area where the repository was established it had previously been a small brook and the area had a slightly inclined surface. A drainage sump was established at the lowest level at one end of the repository.

Retrieval of waste drums

The area around the repository was fenced off and classified as a controlled area where work required use of special clothes and boots and special radiation protection measures were required. The fence had two openings, one for transportation of drums out of the area and one personnel entrance. The personnel entrance was through a tent with a barrier in the middle. The barrier marked the border to the controlled area. The tent was used for changing clothes on entrance to and exit from the restricted area and was also used to store radiation protection instruments and other radiation protection equipment for use during work and stay in the area.

The repository was uncovered in smaller sections and an excavator was used to lift the drums up and out of the repository. The process is shown in Figure 1. Bigger lumps of clay were removed from the surface of the drums before they were lifted out of the repository. A rough removal of clay from the drum surfaces was performed before the drums were repacked. This clay was measured and

Figure 1. An excavator was used to lift the drums out of the repository.

contaminated clay was treated as radioactive waste and collected in two waste drums. Many of the drums were rather heavily stuck in the clay bed and the lifting process put a heavy strain on these drums. One drum cracked and fell apart and the content fell out before it could be removed from the repository. The dose rate from the content (plastic pipes etc.) was however low, and it could be recovered and secured without any problems.

The condition of the drums varied from almost complete intact to heavily corroded with cracks and openings. Figure 2 shows some examples of the condition of drums. It was not necessary to clean the surfaces completely because all the drums should be repacked in new drums.

A major part of the drums were difficult to identify because the marks and signs were either difficult to read or destroyed by corrosion. However, the drums containing plutonium which according to the governmental decision in 1994 should be placed in the storage part of the new KLDRA-Himdalen facility, could be identified by two bungholes compared to the other drums having only one bunghole. A total number of 166 drums containing plutonium were recovered and given new identification numbers. These 166 drums were repacked in new 330 l stainless steel drums. The 831 drums without plutonium and with one bunghole were also given new identification and were designated for depositing in the KLDRA-Himdalen facility. These drums were repacked in new 300 l steel drums.

In order to give the old drums a protective layer and fix the contamination on the surfaces concrete was pumped into the space between the old drum and the

Figure 2. Drums from the repository

new outer drum. The drums were then cleared for transportation and brought to a nearby storage building awaiting transportation to KLDRA-Himdalen. Twenty drums, 10 with plutonium and 10 without, have been selected for further research activities and are stored separately at IFE

Since many drums were impossible to identify the old documentation could not be used to identify the content. The knowledge of nuclides and the total radioactivity level of nuclides in the repository was, however, available and the mean radioactivity per drum could therefore be calculated and was used for drums where identification was impossible.

Radiation protection measures and radiation doses

Monitoring of contamination in air and doses inhalation of radioactivity

People working inside the controlled area were classified as occupationally exposed. Individual doses to occupational exposed personnel were monitored by personal dosimeters based on TL-dosimetry from IFE's radiation protection service. In addition to the personal dosimeters work inside the controlled area required use of electronic personal dosimeters (EPD). Doses on the EPD's were recorded every day after work inside the controlled area. During the period starting at 14 August and ending at 22 October doses to 21 persons were recorded on EPD's during work in the controlled area. The maximum total dose recorded on the EPD's was 2.06 mSv. The dose recorded on the personal dosimeter for the same person for the period July/August – October was 1,8 mSv. The collective dose recorded by the EPD's for the 21 persons was 5.2 man-mSv.

During the excavation work the contamination of the air by radioactive dust particles was monitored by use of a transportable air monitor. The activity on the filter was measured every day with a contamination monitor in order to calculate the level of air contamination. The filter was changed once a week. A similar instrument located at the other end of IFE's premises approximately 400 meters away monitored the background level of air contamination.

Workers that frequently were in contact with drums and clay were equipped with personal air samplers (PAS). Filters in these instruments were changed once a week and the activity on the filters were measured.

Based on the daily and weekly measurements of filters from the air monitor and the PAS' it was decided that use of breathing masks were not required during the work.

Based on g-measurements and radiochemical analysis of the radioactivity on the filters from the air monitor located inside the controlled area and from the air monitor for background measurements the maximum committed dose from inhalation of contaminated dust was calculated. It was assumed that a person worked 5 days and 8 hours a day for a period of 10 weeks. A breathing volume of 20 l/minute was used. The results of the calculations were:

- Maximum committed dose from ^{137}Cs: $< 5\times10^{-4}$ µSv
- Maximum committed dose from ^{238}Pu: $< 5\times10^{-3}$ µSv
- Maximum committed dose from $^{239,240}Pu$: < 0.2 µSv

Dose rates in the environment outside the controlled area

Once a week the dose rate was measured at the gate to the controlled area, at the fence around the nearby kindergarten, along the walls and at a door to the

Table 2. Dose rates in the environment outside the controlled area.

Location	Range of dose rate measurements (µSv/h)
At the gate to the controller area	0.1 – 1.0
At the fence to the kindergarten	0.1 – 0.15
At the wall and door to the nearby building	0.25 – 7.0
At the tent for entrance/exit	0.2 – 1.5

nearby building and at the tent for personnel entrance and exit. The results are shown in Table 2. A TL-dosimeter was placed at the gate for transportation of drums out of the controlled area in the period 2001.09.09 – 2001.11.07. The mean dose rate at the gate in this period was 0.19 µSv/h. The dose rate in the environment depended on the number of drums stored above the ground in the controlled area.

Measurements and free classification of the clay bed

Clearance levels for contaminated clay was specified by the Norwegian Radiation Protection Authority (NRPA) to be:

- For ^{137}Cs : 100 Bq/g dry weight
- For ^{239}Pu + ^{240}Pu + ^{241}Am : 10 Bq/g dry weight

After a section of drums in the repository had been removed samples of clay from the clay bed in 2 x 2 meter grid and from the sides were taken and analysed by γ-spectroscopy. A total of 99 samples were taken from the empty repository. A calibrated MiniSPEC Survying Gamma Ray Spectrometer form Exploranium was used to measure ^{137}Cs in wet clay samples. The detection limit was 10 Bq/g. Calculation of Pu + Am was made by using a relation of 0.08 between Pu + Am and Cs. This relation includes a factor of 2 between wet weight and dry weight. From earlier measurements in 1993 and 1994 it could be concluded that the relation between Pu+Am and Cs content in the clay was below 0.04 with 95 % confidence.

Based on measurements and calculations it could be concluded that all 99 samples taken form the empty repository were below the clearance levels given by NRPA and the repository could therefore be closed without removal of contaminated clay.

Closing of the empty repository

Since 996 drums had been removed from the repository additional masses had to be filled in the empty hole. As a result of the removal of contaminated sediments from the river Nitelva at the end of IFEs old pipeline for release of low-level liquid radioactive waste, 37 containers with approximately 160 - 170 m^3 of sediments were stored at IFEs premises at Kjeller. By measurements it could be established that the radioactivity content in these containers were below the clearance levels given by NRPA. IFE was therefore allowed to fill these sediments into the empty hole together with the layer of soil from the top of the drums that had been removed.

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Development of a simple procedure for the analysis of Pu, Am, Cm and Sr in low-level liquid radioactive effluents

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Abstract

A new sequential procedure for the analysis of Pu, Am, Cm and Sr in low-level liquid radioactive effluents is presented. The proposed method utilises an actinide specific extraction chromatographic resin (TRU-Resin) for the isolation and separation of the actinides. ^{90}Sr determination is accomplished via analysis of its progeny ^{90}Y, which is purified with TBP extraction. The proposed method is simpler and faster than traditional ion exchange, extraction and precipitation methods used in routine analysis of actinides and ^{90}Sr. The chemical yield of all analytes, when analysing 200 ml low-level effluent, is between 80-95%.

Introduction

The operation of nuclear power plants undesirably leads to the formation of liquid and solid radioactive waste containing activation and fission products. While the solid waste is suitably stored in repositories, the low-level liquid waste is normally discharged to appropriate surroundings, after decay of the shortest living radionuclides. Before such an operation can be undertaken the radionuclide content of the liquid must be examined. The Environmental Monitoring Section (EMS) of The Institute for Energy Technology (IFE), performs surveillance of discharges originating from IFE's nuclear activities to ensure that the discharge limits given by the Norwegian Radiation Protection Authority (NRPA) are not exceeded. The liquid effluents are analysed for gamma emitters (including ^{241}Am), ^{3}H, ^{35}S, ^{90}Sr, ^{238}Pu and ^{239,240}Pu. The gamma analysis is performed directly on the liquid samples using HPGe detectors with low-energy windows. Analysis of ^{90}Sr, ^{238}Pu and ^{239,240}Pu is performed using standard separation procedures.

The low-level liquid waste at IFE originates from the operation of a small (thermal output 2 MW) research reactor and the production of radiopharmaceuticals. The waste from different locations at IFE is collected in large drums to allow the decay of the shortest living radionuclides before an evaluation on the discharge of the liquid is undertaken. Representative samples from the drums are then analysed for α-, β- and γ-emitters. The composition of the liquid and the radionuclide content varies with the activities at the specific time.

In our lab ^{90}Sr is analysed using the traditional "fumic nitric acid method" [1,2]. $^{239,240}Pu$ analysis is done on a separate sample using traditional anion exchange procedure before the activity on electroplated discs is determined using PIPS detectors.

In an effort to optimise these analyses the use of extraction chromatography coupled with liquid-liquid extraction to develop a sequential procedure for the analysis of ^{90}Sr and α-emitters was studied. Since the gamma detection limit of ^{241}Am is quite high, an alpha analysis of ^{241}Am was also desired. The wastewater is normally stored for at least a month before radiochemical separations are undertaken. ^{90}Sr activity in the water therefore equals the activity of its progeny ^{90}Y. In the proposed procedure the more favourable separation chemistry of Y(III) is utilized to determine the activity of ^{90}Sr via ^{90}Y. An additional advantage in analysing ^{90}Y is its short half-life ($t_{1/2}$ = 64.1 hours), which enables a simple check of chemical purity of the final source by just following its decay.

Experimental

Reagents and apparatus

Seven different low-level liquid waste solutions were analysed for Sr, Pu, Am and Cm. The ^{242}Pu (0.163 Bq/ml) and ^{243}Am (0,086 Bq/ml) standards were supplied by Risø National Laboratory, Denmark. Y-carrier (10 mg/ml) was prepared from pa grade $Y_2(NO_3)_3 \times 6\ H_2O$ purified by TBP-extraction [1]. The extraction chromatographic resin used for the separation of Pu, Am and Cm was TRU-Resin SPS (50-100 µm) (Eichrom Ind.). All chemicals were of analytical reagent grade.

An i.d. 5 mm glass column with a glass wool fitting at bottom and top was used to make an extraction chromatographic column of 0.50 gram TRU-Resin. 0.1 µm polypropylene membran filters (25 mm) fitted in a 50 ml polysulfone filter funnel (Gelman Sciences) were used to collect the fluorides when preparing the α-sources.

^{90}Y-beta activity was determined using a low level beta GM multicounter (Risø GM-25-5). The alpha-activities were measured with 450 mm^2 PIPS detectors placed in multichamber racks (Canberra). The vacuum in the chambers was kept under 10 torr and the distance between the source and the detector at 1 cm, giving a counting efficiency of 17±1 %. Data treatment was done with the Canberra Alpha Analyst software.

Procedures

200 ml effluent water was acidified with 10 ml 65% HNO_3 and 0.2 ml each of ^{243}Am- and ^{242}Pu-tracer and 1ml of Y-carrier solution added before the sample was evaporated to dryness. The salts were then treated twice with 10 ml 65% HNO_3.

The salts were dissolved with 15 ml 3 M HNO_3 and Na_2SO_3 added to 0,2 M. The solution was left for 20 minutes for the complete reduction of Pu to Pu(III). $NaNO_2$ was added to 0,2 M and the solution left for 20 min for the oxidation of Pu(III) to Pu(IV). The solution was then loaded on a TRU-Resin column pre-treated with 3 M HNO_3 and the column washed with 10 ml 2 M HNO_3 and 10 ml 2 M HNO_3-0,1 M $NaNO_2$ (Flow: 1-2 ml/min). All effluents were kept for Y-analysis. After washing the column with 2 ml 9 M HCl, Am was eluted with 10 ml 4 M HCl and Pu with 10 ml 4 M HCl-0,02 M $TiCl_3$ directly into two different 20 ml scintillation vials. The Am and Pu-fractions were added 50 µg Ce and 1 ml 40% HF and the fluorides allowed to develop for at least 30 min. The fluorides were then collected by filtration through 0,1 µm membrane filter pre-treated with 80% ethanol and the vials and filter washed with 5 ml H_2O followed by 5 ml 96% ethanol. The filters were then attached to planchets with double sided tape, dried at 60 ^{o}C, and the alpha-activity determined using PIPS detectors.

The Y-fraction from TRU-Resin was evaporated to dryness, treated once with 5 ml 65% HNO_3 and the salts picked up with 30 ml 14 M HNO_3. Y was then extracted twice with 30 ml TBP preconditioned with 14 M HNO_3 and the separation time between Sr and Y noted. The TBP phases were combined, washed with 30 ml 14 M HNO_3, and Y stripped twice with 30 ml H_2O. NH_3 was then added to pH 2-3 and Y-oxalate precipitated with the addition of saturated oxalic acid solution. The solution was heated for approximately 1 hour at 90 ^{o}C, cooled, and the oxalate collected on a 25 mm Whatman 42 ash less filter paper. ^{90}Y-activity was then determined using a GM-counter. After β-counting the Y-oxalate was ashed at 900 ^{o}C for 90 minutes and the Y_2O_3 picked up with 3 ml 65% HNO_3. The solution was evaporated to dryness and the salts dissolved with 20 ml acetate buffer at pH 4,4. 1 drop of 0.5% xylene orange indicator was then added and Y titrated with 0.01 M EDTA until a color change from red to yellow was observed. Y-recovery was then determined by comparing the EDTA consumption of the analyte solution versus that of a reference solution containing 10 mg Y [4].

Results and discussion

In normal routine operations where the analysis of Sr and actinides is desired for a sample the analyses are performed on different sample aliquots. One aliquot is analysed for Sr and another for Pu and Am. This requires the pre-treatment of two different sub samples. Since the pre-treatment step for many matrixes is the most time consuming step a considerable gain in throughput can be achieved by using the same sample for both Sr and actinide analysis. In a two-step sequential procedure the time used for the analysis of the second analyte can additionally be reduced as many interfering elements are eliminated in the first step.

In the proposed procedure tri-, tetra- and hexa-valent actinides are extracted by TRU-Resin while most other elements pass right through. Some Y(III) is also extracted, but this is washed out of the column with 2 M HNO_3 [3]. When the column is washed with HNO_3 containing $NaNO_2$ any remaining traces of Pu(III) are oxidised to Pu(IV). Since trivalent actinides are not extracted by CMPO (the active extractant in TRU-Resin) from HCl soluitons, 4 M HCl is used to elute Am and Cm. As Cm(III) resembles Am(III) the recovery of ^{243}Am equals the recovery of Cm and the activity of other Cm isotopes can also be calculated. After the removal of trivalent actinides plutonium can be reductively eluted with 4 M HCl – 0,02 M $TiCl_3$. The α-sources are then made by micro co-precipitation with CeF_3. The alpha spectra of the Am and Pu fraction can be seen in Figure 1. As seen both fractions are very clean and do not contain any interfering α-emitters.

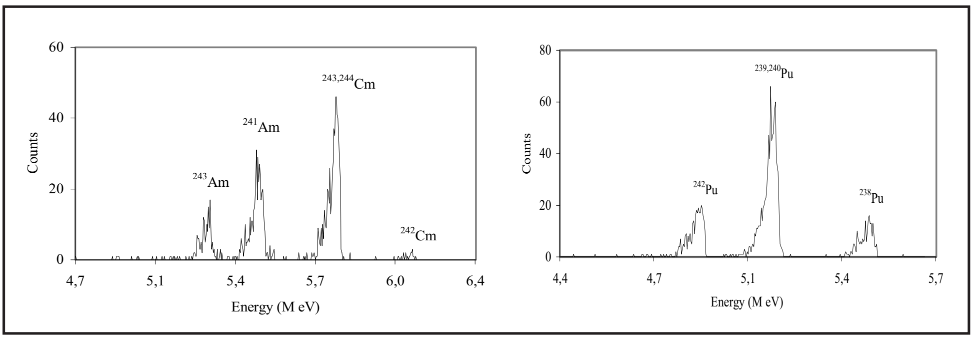

Figure 1. Alpha-spectra of the americium (a) and plutonium (b) fraction.

Table 1. Chemical yields of Pu, Am, Y and Sr using proposed and traditional methods.

Element	Pu	Am	Y	Sr
Yield (Proposed method)	94 ± 9%	87 ± 12%	79 ± 7%	n.d.
Yield (Traditional methods)	24 ± 12%	n.d.	80 ± 12%	67 ± 27%

TBP extraction is here used to extract Y and separate it from Sr and other mono and divalent cations. Only those elements that are not extracted by CMPO from 2-3 M HNO_3 but extracted by TBP from 14 M HNO_3 and form sparingly soluble oxalates at pH 2-3 can interfere in the determination of ^{90}Y. The possible co-precipitation of radionuclides interfering in the determination of ^{90}Y can easily be checked by following the deacy of the Y-source. In none of the seven samples analysed did the decay deviate from that of ^{90}Y. The chemical yield of Y is traditionally checked by gravimetric methods. This however can give a too high yield estimate as other elements also can precipitate together with Y-oxalate. In our lab we have found pH-controlled EDTA titration to be more reliable than gravimetric methods for the determination of Y-recovery [4]. The chemical yield of Pu, Am, Y and Sr for the proposed and traditional methods is shown in Table 1.

While the recoveries of both Sr and Y are high using the tradition method, the recovery of Pu is very low. Using the proposed method the recoveries of all analytes are very high. In addition the new method allows a simple α-analysis of Am and Cm. The traditional method requires the use of about one week for Pu analysis and 3 weeks for Sr-analysis. The proposed method offers significant time reduction as separation and source preparation of Pu, Am and Y can be achieved in one day.

Conclusion

A simple, rapid and reliable method for the analysis of Pu, Am, Cm and Sr in low-level liquid effluents is proposed which offers higher chemical yields, is less time consuming and more environmentally friendly than traditional methods.

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Measurements and calculations for determination of discharge of ^{41}Ar from IFEs research reactor JEEP II at Kjeller, Norway

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Abstract

^{41}Ar is formed by neutron irradiation of ^{40}Ar, which occurs naturally in air with a concentration of 9300 ppm. The discharge of ^{41}Ar from IFEs research reactor Jeep II is yearly reported to the Norwegian Radiation Protection Authority (NRPA). Until year 2000 the reported values were based on theoretical calculations of produced ^{41}Ar per operating hour of 6.8 GBq/h. During 2000 and 2001 the reactor was upgraded to increase the irradiation capacity and to meet the markets demand for irradiation of 5"Si-crystalls. After the upgrading, measurements and calculations were initiated to determine the new discharge rate for ^{41}Ar. During reactor operation an approximately constant discharge of ^{41}Ar is expected, mainly due to irradiation of air in open beam channels. In addition ^{41}Ar is released from irradiation pockets when they are opened to transfer samples in and out during reactor stop. The new value for discharge rate was determined from measurements of air samples from the discharge channel during operation and theoretical calculations of the release from the irradiation pockets. The new discharge rate was determined to 5.9 ± 0.5 GBq/h, which is a small reduction compared to the former value of 6.8 GBq/h. A small reduction in discharge rate was expected because the number of air-filled irradiation pockets was reduced after the upgrading. In a normal year the discharge of ^{41}Ar will be about 2 % of the Institutes discharge permission.

Introduction

During 2000 and 2001 IFEs research reactor Jeep II was upgraded to increase the irradiation capacity and to meet the markets demand for irradiation of 5" Si-crystalls. Larger and water-filled irradiation pockets were installed, and the top lid and various help systems were changed. Until year 2000 the reported discharge was based on theoretical calculations of produced ^{41}Ar per operating hour. The calculated discharge rate was 6.8 GBq per working hour at a reactor effect of 2 MW [1]. Two non-calibrated gamma monitors control the discharge rate, and give alarm on high activity levels [2]. After the upgrading, measurements and calculations were initiated to determine the new discharge rate for ^{41}Ar. A small reduction in discharge rate was expected after the upgrading because the number of air-filled irradiation pockets had been reduced.

Sources of ^{41}Ar discharge

^{41}Ar is formed by neutron irradiation of ^{40}Ar, which occurs naturally in air with a concentration of 9300 ppm. During reactor operation an approximately constant discharge of ^{41}Ar is expected, mainly due to irradiation of air in open beam channels. In addition ^{41}Ar is released from irradiation pockets when they are opened to transfer samples in and out during reactor stop.

The reactor is equipped with 9 irradiation pockets, 2 which are filled with air and 7 which are circulated with heavy water. In the water-filled pockets ^{41}Ar is produced by irradiation of ^{40}Ar that is dissolved in the water. The irradiation pockets are closed during reactor operation by a plug on the top. The plug is removed when transferring samples in and out of the pockets during reactor stop, and the ^{41}Ar that has been created during operation is released. In addition to the radiation pockets described above, a tenth pocket contains a pipe system called the “Rabbit” in which samples are transported to and from the reactor core for neutron activation. ^{41}Ar created in the “Rabbit” is released when transferring samples in and out of the “Rabbit”, which is usually done during reactor operation.

Measurements of ^{41}Ar from the beam channels

The discharge rate from ^{41}Ar produced in the beam channels was determined by measurement of air sampled from the discharge channel. Measurements were assumed to be easier and more accurate than calculations, mainly because accurate stipulations of the parameters needed for calculations are difficult. Measurements also make it possible to control whether the discharge rate is constant with time.

Air was sampled from the discharge channel on 1-litre evacuated glass flasks. The samples were measured on a Canberra 3x3” NaI-detector. Efficiency calibration of the detector was performed using a standard ^{60}Co-solution. Corrections were made for differences in self-attenuation in air and water.

The reactor is usually started and stopped several times during a day to

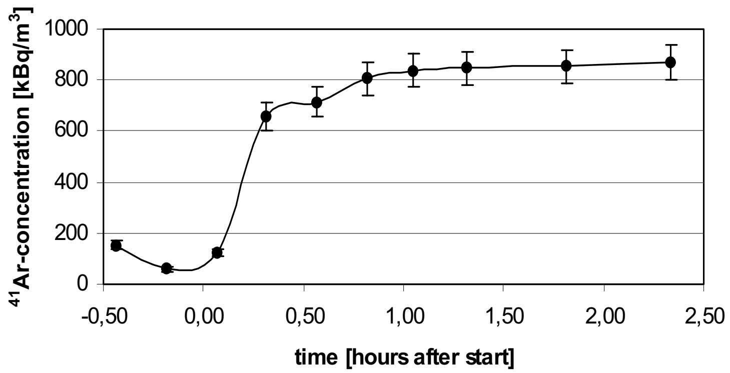

Figure 1. Samples collected during and after starting of reactor.

transfer samples in and out of the irradiation pockets. After the reactor is started, some time will elapse before the ^{41}Ar concentration at the sampling place is stabilized at a maximum level. This was examined by taking a series of samples during and after start of the reactor. The results are shown in figure 1. After about one hour the ^{41}Ar concentration was stabilized on a maximum level of 850 kBq/hour.

In order to determine the mean discharge rate during reactor operation, several samples were collected at arbitrary hours during reactor operation. The sampling was done after the reactor had been operating on maximum effect (2 MW) for at least one hour. Two parallel samples were collected. The results are given in Table I.

Except for the samples collected September 12th it was good agree-ment between the concentrations measured in the two parallel samples. Possible reasons for the day-to-day differences in concentrations may be uneven air stream and air density.

Table I. ^{41}Ar concentration in samples taken at arbitrary times during reactor operation.

Date	Concentration [kBq/m^3]	
	Sample 1	**Sample 2**
05.09.01	771	771
10.09.01	840	848
11.09.01	776	763
12.09.01	900	777
14.09.01	674	670
19.09.01	900	906
All samples:	***800 ±80***	

The discharge rate during reactor operation is:

$$U = \overline{u}_{luft} \cdot \overline{c} \qquad \text{Formula 1}$$

where

U	=	Discharge rate in Bq/h
$\overline{u}_{luft}$	=	Average discharge rate for air in m^3/h
$\overline{c}$	=	Average activity concentration in Bq/m^3

The average discharge rate for air was measured by the Building and Maintenance department to 6100 ± 100 m^3/h. The resulting discharge rate based on the samples collected is

4.9 ± 0.5 GBq/h.

Calculations of contribution from irradiation pockets

The determination of the discharge rate from the irradiation pockets was based on calculation of the amount of ^{41}Ar produced in the pockets during reactor operation, and knowledge of how often the pockets are opened. For the air-filled pockets it was assumed that all the produced ^{41}Ar is released when opening the pocket. The water-filled pockets are circulated with heavy water, and ^{41}Ar is produced by activation of ^{40}Ar dissolved in the water. It was assumed that all the ^{41}Ar dissolved in the water that passes the opening of the pockets when the pockets are open is released.

The activity of the ^{41}Ar produced by neutron activation of ^{40}Ar after an irradiation time t is given by:

$$Q = \frac{\varphi \cdot q \cdot f \cdot N_A \cdot \sigma}{A_m} \cdot V(1 - e^{-\ln 2 \cdot \frac{t}{T_{1/2}}}) \qquad \text{Formula 2}$$

where

Q	=	activity in Bq
φ	=	thermal flux in $cm^{-2}\ s^{-1}$
q	=	concentration of Ar in g/m^3 (= 16.8 g/mol in air)
f	=	relative amount of ^{40}Ar = 0.996
N_A	=	avogadros number = $6.023 \cdot 10^{23}$ mol^{-1}
σ	=	cross section for $^{40}Ar(n_t, \gamma)^{41}Ar$ = 0.64 barn = $0.64 \cdot 10^{-24}$ cm^2
A_m	=	molar weight for Ar = 39.948 g/mol
V	=	irradiated volume (m^3)
t	=	irradiation time
$T_{1/2}$	=	half life for ^{41}Ar = 1.83 h

Air-filled irradiation pockets

The reactor has two air-filled irradiation pockets, I6 and E2. Table II shows the parameters for irradiated volume, thermal flux and the calculated activity of ^{41}Ar. It is assumed that the irradi-ation time is long enough so that equilibrium is reached. Thermal flux measured in the height of the active part of the fuel elements varies from 4.0×10^{12} – 9.0×10^{12} $cm^{-2}s^{-1}$ in E2, and from 8.5×10^{12} – 1.5×10^{13} $cm^{-2}s^{-1}$ in I6 [4]. It is assumed in the calculation that all the air in the pocket is irradiated with the maximum thermal flux, which is a very conservative assumption taking into account that the active part of the fuel elements only covers about one third of the total height of the pockets. It is also assumed that the pockets are empty. Most of the time the pockets will be filled with samples, which displaces about 10 % of the total volume. The resulting activities are therefore considered to be conservative. The pockets are on average emptied three times a week [4]. Average discharge rate from the air-filled pockets is then:

Discharge rate = $Q_{total} \times 3/(24\text{ h} \times 7)$ = <u>0.81 GBq/h</u>

Table II. Argon in air-filled irradiation pockets.

Pocket	V [m^3] [5]	φ [cm^{-2}s^{-1}]	Q [Bq]
E2	0.0117	$9.0 \cdot 10^{12}$	$1.7 \cdot 10^{10}$
I6	0.0117	$1.5 \cdot 10^{13}$	$2.8 \cdot 10^{10}$
		Total:	***$4.5 \cdot 10^{10}$***

Water-filled irradiation pockets

The reactor has seven water-filled (heavy water) pockets, five of type I, and 2 of type E. During operation ^{41}Ar is produced by irradiation of ^{40}Ar dissolved in the water. The solubility of Ar in the water was calculated using Henrys law:

$$Solv = k \cdot P \qquad \text{Formula 3}$$

where

Solv	=	Solubility of gas in mol/litre
k	=	Constant = $1.4 \cdot 10^{-3}$ M atm^{-1} at 25 °C for Ar
P	=	Partial pressure of the gas above the liquid = 0.00934 atm for Ar in air

This gives a solubility of 1.3×10^{-5} mol Ar per litre water.

Because the water is circulated through the pockets, not all of the water is irradiated at the same time. Due to the short half-life of ^{41}Ar, the produced ^{41}Ar in a water volume will disintegrate when circulating outside the pocket until the same volume is irradiated once more. Total volume, volume irradiated at one time, and neutron flux is given in table III.

Table III. Parameters used for calculation of ^{41}Ar activity.

Pocket	Irradiated volume per pocket [m^3]	Total volume per pocket [m^3]	φ [cm^{-2}s^{-1}]
Type E (2)	0.0177	0.0583	$9.0 \cdot 10^{12}$
Type I (5)	0.0177	0.0583	$1.5 \cdot 10^{13}$

The irradiation time was calculated to be 0.6 h, and the disintegration time to 1.4 h based on a water flow of 0.03 m^3/h. In table IV the same finite water volume is followed through 6 irradiation and disintegration periods in an E-pocket. The ^{41}Ar concentration at the end of each period was calculated from Formula 2. After 4 periods the concentrations has stabilized at 1.02×10^{10} Bq/m^3 at the end of an irradiation period and 6.16×10^{9} Bq/m^3 at the end of a disintegration period. Equivalent calculations were done for the I-pockets.

Table IV. ^{41}Ar concentration in an infinite water volume in an E-pocket at the end of each irradiation and disintegration period.

Time [h]	Activity concentration [Bq/m^3]		Time [h]	Activity concentration [Bq/m^3]	
0.59	$9.04 \cdot 10^{9}$	1. irradiation	6.42	$1.02 \cdot 10^{10}$	4. irradiation
1.94	$5.42 \cdot 10^{8}$	1. disintegration	7.78	$6.16 \cdot 10^{9}$	4. disintegration
2.53	$1.01 \cdot 10^{10}$	2. irradiation	8.36	$1.02 \cdot 10^{10}$	5. irradiation
3.89	$6.05 \cdot 10^{9}$	2. disintegration	9.72	$6.16 \cdot 10^{9}$	5. disintegration
4.48	$1.02 \cdot 10^{10}$	3. irradiation	10.3	$1.02 \cdot 10^{10}$	6. irradiation
5.83	$6.10 \cdot 10^{9}$	3. disintegration	11.7	$6.16 \cdot 10^{9}$	6. disintegration

The pockets are opened twice a day in average, i.e. every 12th hour. It may therefore be assumed that the ^{41}Ar concentration is at a maximum level (table IV) when the pockets are opened. It is assumed that the pockets stay open for 10-15 minutes when opened to transfer samples in and out. After 15 minutes 0.0075 m^3 water has passed the opening of the pocket (15 cm diameter). It is assumed that the water passes the opening after being irradiated, and then has a maximum ^{41}Ar concentration (1.02×10^{10} Bq/m^3 for E-pockets and 1.71×10^{10} Bq/m^3 for I-pockets). Assuming that all the ^{41}Ar in the 0.0075 m^3 that passes the opening of the pocket is released, 7.7×10^{7} Bq is released from each E-pocket every time it is opened. The corresponding value for an I-pocket is 1.3×10^{8} Bq. The average discharge rate is then:

Discharge rate = ($2\times7.7\times10^{7}$ Bq + $5\times1.3\times10^{8}$ Bq)×2/24 h = 0.07 GBq/h.

A considerable part (up to 45 %) of the water available for irradiation is usually displaced by crystals. The calculated discharge rate is therefore assumed to be conservative.

"Rabbit"

^{41}Ar created in the "Rabbit" is released through the discharge channel when samples are transferred in or out of the "Rabbit". The amount of ^{41}Ar created in the "Rabbit" is calculated by Formula 2. It is assumed that the irradiation time is long enough so that equilibrium is reached. Volume of irradiated air, neutron flux and calculated ^{41}Ar activity is given in table V. The "Rabbit" is in average used 4 times a week [3], and ^{41}Ar is released both when taking samples in and out of the "Rabbit". Average discharge rate is then:

Discharge rate = $Q_{total} \times 4/(24\ \text{h} \times 7)$ = <u>0.15 GBq/h</u>

Some of the air available for irradiation is usually displaced by samples. The calculated discharge rate is therefore assumed to be conservative.

Total discharge rate

The average discharge rate for ^{41}Ar from JEEP II is given in table VI. The total yearly dis-charge is found by multiplying the discharge rate by the number of operational hours for the reactor.

Because the calculated contributions from the irradiation pockets are regarded as being con-ser-vative, no uncertainty is given for these values.

Discussion and conclusion

Before upgrading of the reactor there were 6 air-filled irradiation pockets. A reduction in dis-charge rate was expected after the upgrading because the numbers of air-filled pockets were reduced to 2. However the "Rabbit" was not included in the former calculations. In addition some release is taken place

Table V. Argon in the "Rabbit".

"Rabbitt"	V [m^3]	φ [cm^{-2}s^{-1}]	Q [Bq]
	0.0038 [6]	$1.0 \cdot 10^{13}$	$6.4 \cdot 10^{9}$

Table VI. Total discharge rate from JEEP II.

Contribution from	U [GBq/h]
Beam channels	4.9 ± 0.5
Air-filled pockets	0.81
Water-filled pockets	0.07
"Rabbit"	0.15
Total:	***5.9 ±0.5***

from the water-filled pockets. The resulting discharge rate of 5.9 GBq/h compared to the former value of 6.8 GBq/h therefore seems reasonable. In 2000 the reported discharge of ^{41}Ar from JEEP II was 22.8 TBq, based on a discharge rate of 6.8 GBq/h, and 3361 ope-ratio-nal hours [8]. This amounts to only 2 % of the discharge permission. With discharge levels far below the discharge permission the precision of the perfor-med measurements and calculations is considered to be good enough, and the reporting of discharge should be based on the discharge rate of 5.9 GBq/h.

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Posters

4a Long Term Behaviour of Waste Drums Retrieved from the IFE-Kjeller Near-Surface Repository

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Abstract
In 1970 a near surface repository containing about 9300 GBq radioactive waste was established at IFE-Kjeller. The buried waste units were retrieved during the autumn of 2001.

15 waste drums had been retrieved from the repository earlier. A pilot study was carried out on one of these drums. The aim of the investigation was to determine if leakage of radionuclides through the barriers had taken place. The conclusion was that the investigated drum was in good condition and no leakage of radioactive material had taken place.

From the drums retrieved in 2001, 20 drums have been selected for further studies. Proposed studies include core sampling and the measurement of activity profiles through the barriers surrounding the waste.

Introduction
In 1970 a total of nearly 1000 drums and 19 other units containing radioactive waste were buried in clay at IFE-Kjeller. After a safety assessment the drums were disposed with approval from the Norwegian Radiation Protection Authority. The drums were stacked in two layers. The field, depicted in Figure 1, had an area of 276 m^2.

The drums were buried in clay. The reason for choosing clay was that it provides a good barrier against leakage of radioactivity from the field.

Water running through the field was collected in a drain sump. The water was controlled before it was released. The annual release of radionuclides from the repository is estimated to less than 0.1% of the permitted release limit.

When the drums were buried in 1970, this method was recommended by the IAEA and in accordance with current international practise. Radiation protection policy has however changed since then. When the Norwegian parliament in 1994 decided to build a combined storage and repository for low- and intermediate level radioactive waste in Himdalen (KLDRA), it was also stated that the drums should be retrieved and transferred to this new repository.

It should also be mentioned that in 1993, representatives from the environmental foundation "Bellona" committed a forced entry to the premises and uncovered some drums, one of which were damaged by the mechanical digger. IFE took advantage of the incident and dug up 5 drums for inspection. In addition, 10 drums were retrieved in 1994.

Inventory

The radioactive waste consisted of laboratory waste, organic liquid absorbed in vermiculite and dried ion exchange mass. Metallic waste was embedded in concrete. For high dose rate waste, the drum was equipped with a lead inner container. The total activity in the repository is calculated to be 2900 GBq (1.7.1997). The activities for the most important nuclides are shown in Table 1.

The repository contains 80 GBq plutonium (35 g) in 166 drums and 1.3 GBq uranium (100 kg) from the former reprocessing pilot plant.

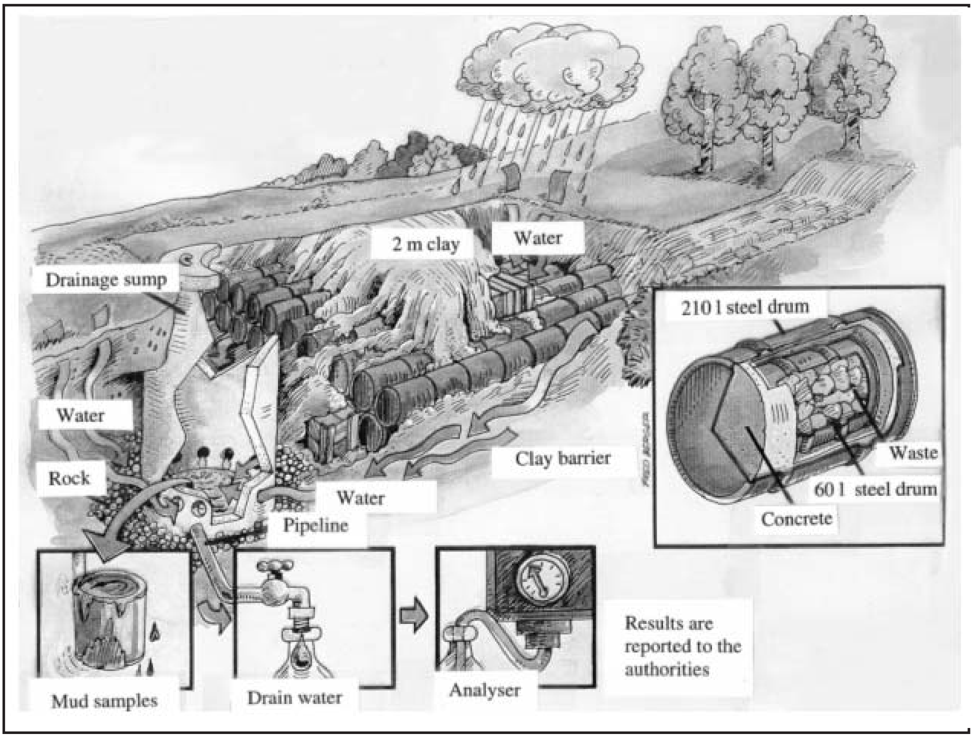

Figure 1. Artistic view of the near surface repository at IFE-Kjeller.

Table 1. Inventory of the near surface repository at IFE-Kjeller

Nuclide	Half-life [years]	Activity [1)] [GBq]
^{60}Co	5.3	60
^{90}Sr	28	1340
^{137}Cs	30	1360
^{239}Pu	24 000	85
U(nat)	$4.5 \cdot 10^9$	1.3
Other [2)]	-	<100

[1)] Activity as per 1. July 1997
[2)] Other nuclides includes ^{3}H, ^{14}C, ^{55}Fe and ^{204}Tl

Contamination in the surroundings of the repository

With two exceptions, no contamination has been detected in the surroundings of the buried drums. One borehole close to the drain sump showed activity concentrations of ^{137}Cs of 10-600 Bq/kg. A trace of ^{137}Cs was also found in another borehole.

Contamination of clay and water in the repository

The activity levels in rust and clay from the drums range from 1.5 MBq/kg and down. Samples of water taken after retrieval of the drums had a content of ^{137}Cs in the range

1-50 Bq/l and for ^{90}Sr from 1-10 Bq/l. Clay and sludge profiles from the bottom of the repository had activity levels of ^{137}Cs up to 100 kBq/kg in the first 2 cm. The concentration of ^{239}Pu in clay was 500 Bq/kg or below with one exception of a sample of 7 kBq/kg. Samples from profiles showed that the activity was concentrated in the 2 cm of clay closest to the drum surface.

Study of the waste drums

Identification and documentation

All drums were initially marked by drum numbers painted on the surface. For each drum number, there exists a drum journal with information on the waste and the type of container. The information content of the journal is, however, of variable value. Some of the drum journals do not contain information on radionuclides present or activity of the waste. After 21 years in the ground, the numbers on many drums were impossible to read. In conclusion, little is known about the activity and isotope contents of each drum. Drums containing plutonium were however easy to identify as the have two bungs.

All drums were given new identification numbers and were photographed before reconditioning.

Pilot study

One of the drums retrieved in 1973/1974 was selected for a pilot investigation. The drum was selected on the criteria that the outer surface was among the most heavily corroded and that information of the composition of the waste was available. The aim of the investigation was to:

- Determine if leakage of radionuclides through the barriers had taken place.
- Inspect the barriers, especially the concrete shielding, for visible degradation.
- Check the information in the journals on drum construction (shielding, inner container).

The method used was to drill out cores from the concrete shielding using a commercially available core-drilling device that generated cores with a diameter of 7 cm (Figure 2). The length of the cores corresponded to the thickness of the concrete shielding, typically 10-15 cm. Three cores were taken from the drum.

The cores have been analysed using high-resolution g-spectrometry. In order to make a measurement of the inner and the outer core end and compare the spectra from the two sides. If there is a difference in activity, the cores were sliced into appropriate samples for a quantitative analysis.

Figure 2. Drilling cores from drum no. 6.

The measurements showed that the samples contained ^{137}Cs. It was, however, no difference in the activity on the two sides. It should also be noted that loose corrosion sampled from the outer surface of the drum had a much higher ^{137}Cs activity.

Measures have been taken to ensure that radiation protection is attended to during the investigation. All treatment of waste drums was performed in an “active area”, that is an area that only can be entered using protective clothing and wearing a personal dosimeter. Effort was made to find a suitable area with acceptable working conditions and where possible contamination from core drilling can be handled safely.

The external dose rate from the selected drum was rather low, typically 20 µSv/h. The main risk was thus connected to contamination. The core drill is cooled using running water, so the risk for generating airborne contamination was low. The spill water may however be contaminated. Care was taken to collect the water in a drip-tray. The amount of water used was kept as low as possible to reduce the amount of liquid waste generated. The idea was to let the water evaporate naturally from the drip-tray.

Protective clothing as rubber gloves and a disposable suit were used during core drilling. A full-protection breathing mask was required, as some cooling water tends to spurt from the drill.

Preliminary conclusion

Visual inspection of the cores revealed that the concrete shielding was in remarkably good condition. No visible degradation or cracks were observed. The inner steel container was also found to be in good condition with no visible corrosion.

The investigated drum was in good condition and leakage of radioactive material had not taken place. It is necessary to investigate more drums to establish if this drum is representative for the majority of drums from the repository. The presence of ^{137}Cs on the outer surface does indeed indicate that some of the drums must have been leaking.

Further studies

Most of the nearly 1000 drums retrieved have already been reconditioned and are being transferred to the KLDRA repository and storage site.

From the drums retrieved in 2001, 20 drums have been selected for further studies. Ten of the drums, easily identified having double bungs, contain plutonium. Proposed studies include core sampling and the measurement of activity profiles through the barriers surrounding the waste.

4b Sorting of waste from the decommissioning of the nuclear facilites at Risø

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Abstract

It has been decided to close down and decommission all nuclear facilities at Risø National Laboratory. The radioactive parts are to be stored in a suitable repository, and the non-radioactive (or very low-radioactive) parts should be carefully checked, before they can be cleared and disposed as ordinary waste. Sorting materials into radioactive and non-radioactive waste is important, as it has both ethical and economical aspects, because the amount of waste to be stored as radioactive waste can be significantly reduced.

Every piece of material removed from the nuclear facilities will be marked and registered, so its movements in the decommissioning system can be followed through the documentation. Due to its original position in the facility, it might have been activated by neutrons, contaminated or both. The flow of material through the sorting system is described in detail, from the first check to the final determination of activity content and the subsequent separation into radioactive and non-radioactive waste.

There are two reasons for determining the radioactivity. One is to have documentation for the activity content in the radioactive waste to be stored and the other is to verify that the activity content is so low that the piece can be disposed off as ordinary waste. The plan is to have two separate laboratories for these measurements.

International organisations have given recommendations on radionuclide specific clearance levels based on a dose rate of 0.01 mSv/y to the most exposed individual. The levels that will be used at Risø will be in accordance with notices given by the Danish authorities. The whole process of clearance, including the measurements of activity content will be described in detailed procedures. The system will be written up as a QA-system, and the plan is to seek certification.

Introduction

The decommissioning includes three research reactors, a hotcell facility and a fuel fabrication plant. The waste treatment plant is also supposed to be decommissioned, but not until a final storage facility is established.

Waste from the decommissioning has to be treated according to strategies laid out by the authorities. Most important in the handling and storing of the radioactive waste is to minimise the impact on people and the environment. Sorting and decontamination will be used to reduce the amount of medium and low-level radioactive waste. The waste will be thoroughly described to fulfil the requirements to the waste storage, the temporary as well as the final.

Sorting of waste

When a part in a nuclear facility is removed from the position is has occupied during operation, it will be marked with an identification. This identification shall contain information on what function the part had, and where it was positioned giving information about possible irradiation with neutrons and possible contamination. The identification shall follow the part through the checking and measuring procedures till it is either disposed off as non-radioactive materials or it is stored in the temporary storage facility. The identification label shall have comments and measuring results added to the basic information. If a part is cut into sub-parts, each sub-part shall have a "daughter"-label. Parallel to the identification label a record with a unique identification number will be created in an electronic database. This database shall contain and link all available information about the different parts.

The purpose of the sorting is to separate the radioactive waste from the non-radioactive waste. The radioactive waste will be placed in standard waste containers to be stored in a temporary storage hall at Risø for later to be transferred to a permanent waste repository. The nuclide specific activity content of the radioactive waste will be documented from the results of the activity measurements and from the chemical analysis of the content of other toxic materials, e.g. heavy metals. The basic principle in the waste sorting is that all waste is radioactive till proven otherwise.

The non-radioactive waste will be recycled or disposed of as ordinary building and metal waste. The activity concentration in this waste will be measured by different methods to demonstrate and document that the concentration is below the clearance level given by the authorities.

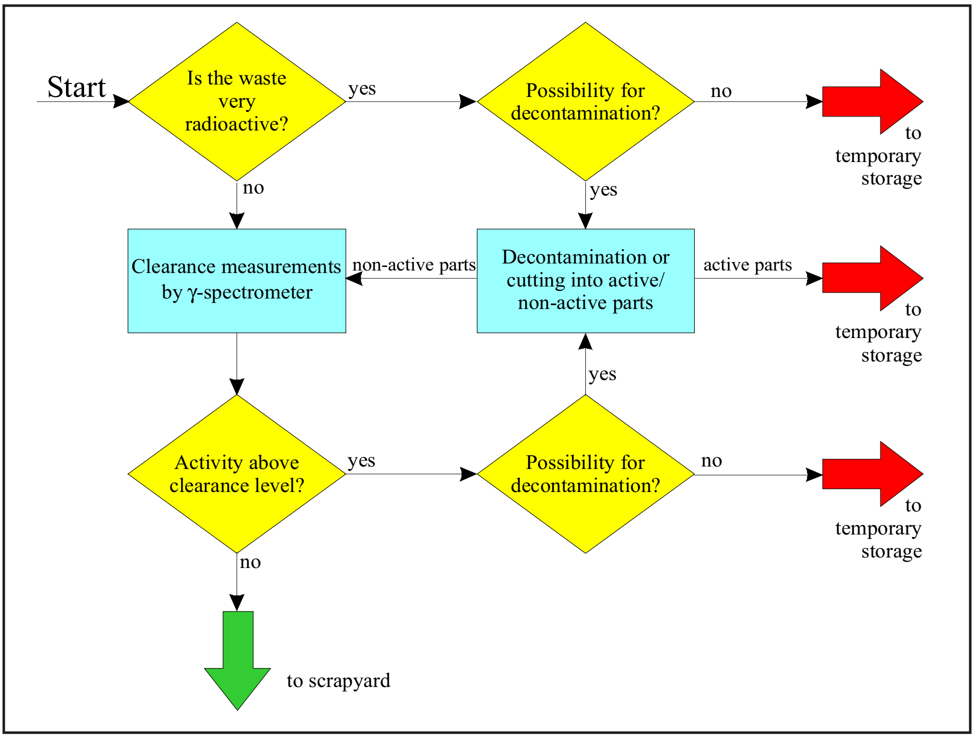

Figure 1. Flow diagram for the sorting of waste from the decommissioning of Risø's nuclear facilities. If the concentration of a radionuclide in the material is above the clearance level, the waste will if possible be decontaminated and measured again - otherwise it will be stored in a temporary storage hall at Risø. If the activity concentration is below the clearance level, the material will be transported from Risø for recycling or for ordinary disposal.

Figure 1 shows the different steps in the sorting procedure for waste from the decommissioning. Materials waiting for measurements will be kept in ISO containers with labels showing which measurements the materials are still needed for the materials. In total 15,000 - 20,000 m^3 of waste is to be handled, about 5,000 m^3 of these are expected to be radioactive and should therefore be stored in the temporary waste storage.

Low-level laboratory for clearance measurements

Control and documentation of the content of g-emitting radionuclides of potentially non-radioactive waste from the decommissioning of the nuclear facilities will primarily be based on gamma-spectrometric analyses. For these measurements a new laboratory will be build. Since is has to be a low-level laboratory, it must be placed at a suitable distance from the nuclear facilities -

particularly DR3 - where highly radioactive materials are handled during the decommissioning. It will also be necessary to classify the laboratory as a clean laboratory; consequently the staff should change to clean and non-active clothes when entering the laboratory.

The laboratory will be supplied with two sets of identical measuring equipment in order to have a reasonable capacity. The object to be checked will be placed on a table. Either the table or the detector must be able to rotate, so that the detector can see all external surfaces. The measuring equipment will be high efficiency germanium detectors. They will be calibrated from a sophisticated point source/volume source technique, which will take into account a possible inhomogeneous activity distribution in bulky items. Furthermore there will be detectors for analysing the content of pure a- and pure b-emitters in samples taken from the items.

All procedures for the measurements shall be approved by the authorities and the laboratory shall be certified according to ISO standards.

Clearance levels

A large part of the waste from the decommissioning will be candidate for release as non-active waste. Non-active waste can without restrictions be disposed outside the Risø area as normal building and metal waste. It is, however, necessary to ensure that its content of activity is sufficiently low, so that no form of post-release regulatory involvement is required in order to verify that the public is being sufficiently protected. The clearance levels is defined by international organisations as *values, established by the regulatory authority and expressed in terms of activity concentrations, at or below which sources of radiation may be released from regulatory control.*

The EU Article 31 Group of experts has made recommendations on the clearance levels for radionuclides in waste from the dismantling of nuclear installations (EU Radiation Protection No. 113 (2000)). These clearance levels have been calculated from public exposure scenarios and a dose criterion of 10 μSv/y, corresponding to what has been defined as a trivial risk. Clearance levels for the radionuclides that are expected during the decommissioning of the nuclear facilities at Risø are shown in table 1.

The content of radionuclides in the candidate waste for release shall be documented to the regulary authorities.

Table 1. Recommended clearance levels from EU.

Clearance levels (Bq/kg)	
^{3}H	10^{5}
^{60}Co	10^{2}
^{63}Ni	10^{6}
^{90}Sr	10^{3}
^{137}Cs	10^{3}
^{238}U	10^{3}
^{239}Pu	10^{2}
^{241}Am	10^{2}

Conclusions

All the nuclear facilities at Risø National Laboratory, except the Waste Management Plant, have been closed and the plan is to decommission these facilities, including the Waste Management Plant, to green field within the next 10 - 20 years. About 15,000 - 20,000 m^3 of waste have to be sorted into non-radioactive waste for release as ordinary waste and into radioactive waste. The latter is estimated to be 5 000 m^3, and it will be stored in standard containers at a temporary storage facility at the Risø area. The activity content in all waste will be determined by different measurements, so that both the released and the stored waste will be well documented.

SESSION 5. EMERGENCY PREPAREDNESS

ICRP publication 82 on protection against prolonged exposure - application in accident situations

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Introduction

In the past 15 years, two events have occurred that cover a conceivable range of emergencies involving extensive post-emergency phase response, namely the Chernobyl and Goiânia accidents. Large amounts of ^{137}Cs were released to the environment during these accidents, leading to a prolonged or quasi-prolonged exposure of the affected populations. The experience gained from these accidents and others have revealed that there is a need for an updated and fully integrated system of guidance on implementation of countermeasures. A revised system must have a sound technical basis but must also be understandable, explainable and acceptable to the public and the decision-makers. Once all protective actions have been undertaken, the situation should be considered 'normal' again with no further restriction being imposed. Therefore a common language explanation should be developed for the public and public officials that clearly state the risks of radiation exposure and what actions are appropriate and inappropriate, and what is "safe". The concepts of "safe" and "return to normality" should be developed together with intervention criteria, disengaged from the linear non-threshold risk hypothesis. Within this context, the application of the recommendations in ICRP Publication 82 for application in post-accident situations is briefly summarized with reference to observations and lessons learned from the Chernobyl and Goiânia accidents.

Radiation protection in prolonged exposure situations

ICRP has recently published guidance on protection of the public against prolonged radiation exposure[1]. Prolonged exposures are adventitiously and persistently incurred by the public over long periods of time. They are incidental to situations in which members of the public may find themselves. The annual doses associated with prolonged exposures are more or less constant or decreases slowly over the years. Generic reference levels for intervention in prolonged exposure situations, expressed in terms of existing annual dose, are recommended by the ICRP; these levels should be viewed as a *consequential* derivation from the basic ICRP principles of radiological protection for intervention and as *complementary*, rather than alternative, to those principles. Their use should not preclude the application of these basic principles to any dose component of the existing annual dose that is controllable, particularly if it is a dominant component.

Sources of prolonged exposure

Situations of prolonged exposure of the public include the prolonged background exposure and the exposure from human-made radiation sources. The prolonged background exposure varies with the geographical and geological characteristics but also with features associated with human development. The human-made radiation sources causing prolonged exposure would arise from a number of human activities associated with the development of society.

The natural sources, which are responsible for the prolonged exposure, are the external cosmic radiation, the radionuclides produced by cosmic rays in the atmosphere (*e.g.* ^{14}C and ^{3}H), and the radionuclides of uranium and thorium in the earth's crust. The exposure pathways include external exposure and inhalation and ingestion of radionuclides in air, food and water.

Prolonged exposures of the public from human activities usually result from releases of long-lived radionuclides into the environment. Residues containing long-lived radionuclides from past human activities that were not adequately controlled are one example. Others are current practices, some past industrial applications, especially mineral extraction, military operations and nuclear or radiological accidents. Figure 1 presents a schematic illustration of various sources of prolonged exposure.

[1] INTERNATIONAL COMMISSION ON RADIOLOGICAL PROTECTION, *Protection of the Public in Situations of Prolonged Radiation Exposure*. Publication No. 82, Pergamon Press, Oxford, New York (2000).

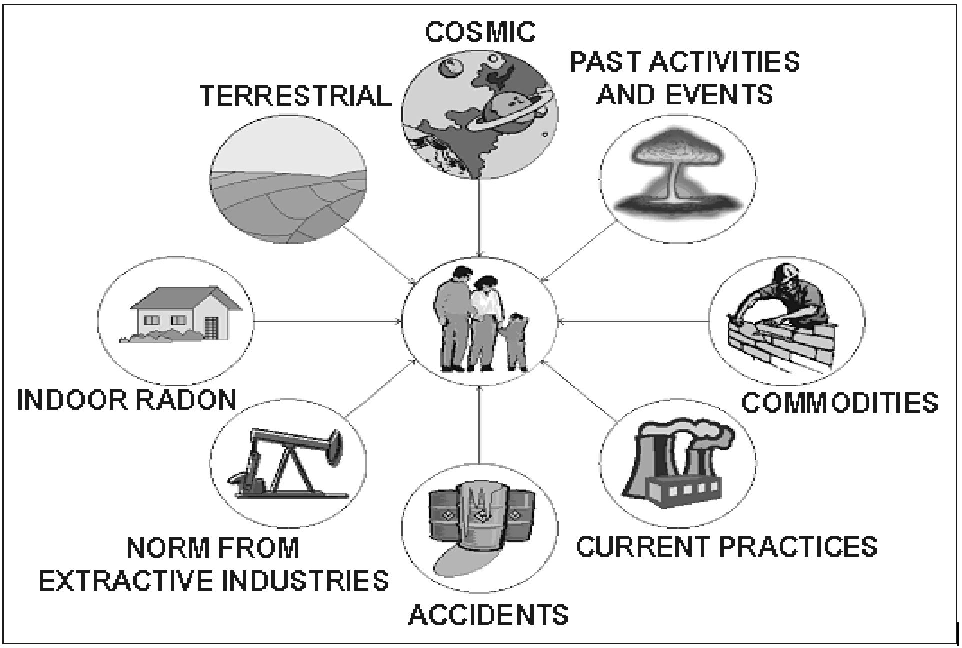

Figure 1. Schematic presentation of various sources of prolonged exposure. They include a number of natural and artificial sources. The sum of exposures to the sources present in a human habitat results in an **existing annual exposure** to the individuals living there.

The operation of practices may leave long-lived radioactive residues in the environment, resulting in situations of prolonged exposure. Practices may also generate prolonged exposure situations due to the disposal of radioactive wastes. Radioactive residues from practices can either result from normal discharges to the environment or remain on and around the site of a practice after the cessation of the practice and decommissioning of its installations.

Intervention situations involving prolonged exposure are of various types. In all cases, decisions have to be taken on whether and how to intervene in order to reduce these exposures and, eventually, on whether and when to discontinue protective actions. The classical intervention situation is where people are already incurring exposures attributable to an identifiable event relatively close in time, *e.g.*, a nuclear or radiological accident. Another type is, for example, exposures to natural sources and to radioactive residues that cannot be linked to any particular originating cause or where the link to the cause has weakened over time.

Reference levels for intervention in prolonged exposure situations
In the ICRP 82 the relevant dosimetric quantity for controlling prolonged exposures is the *annual effective dose*. This is the sum of the time integral, over a year, of the effective dose rate due to external irradiation caused by the prolonged exposure situation and the committed effective dose due to internal contamination caused by all intakes, during that year, of the long-lived radionuclides (and their short-lived progeny) involved in the situation. A subsidiary quantity used within the context of prolonged exposure is the *existing annual dose* caused by all persisting sources of prolonged exposure in a given situation. Other subsidiary quantities are the *additional annual dose* caused by practices and the *averted annual dose* precluded by an intervention.

The existing annual dose can conceptually be used to establish *generic reference levels* for intervention. However, such quantity should be used with caution. It is made up of all the existing and persisting annual doses incurred by individuals and, therefore, it is constituted by many different components of prolonged exposure. These include external exposure to long-lived radionuclides (and their progeny) in soils, strata, and building materials (including exposure to radon and other radionuclides in the ambient), internal exposure due to the incorporation of those radionuclides into the body as a result of inhalation of resuspended materials, and ingestion of contaminated foodstuffs.

There is not a single measure that can be used to determine the value of the existing annual dose, as any of its components may require different assessment methodologies. Thus, there may be practical problems in implementing regulatory standards expressed in terms of the existing annual dose. Because of these difficulties, the ICRP 82 has given preference to *specific reference levels* based on avertable annual doses of given components, rather than to generic reference levels based on existing annual doses. The ICRP 82 recommends that:

(a) *An existing annual dose approaching about 10 mSv may be used as a generic reference level below which intervention is not likely to be justifiable for some prolonged exposure situations.*

(b) *Below the level of existing annual dose for which intervention is not likely to be justifiable, protective actions to reduce a dominant component of the existing annual dose are still optional and might be justifiable. In such cases, action levels specific to particular components can be established on the basis of appropriate fractions of the recommended generic reference level.*

(c) *Moreover, above the level of existing annual dose for which intervention is not likely to be justifiable, intervention may possibly be necessary and its*

justification should be considered on a case-by-case basis as appropriate.
(d) *Situations in which the annual (equivalent) dose thresholds for deterministic effects in relevant organs could be exceeded should require intervention.*
(e) *An existing annual dose rising towards 100 mSv will almost always justify intervention and may be used as a generic reference level for establishing protective actions under nearly any conceivable circumstance.*

In general, ICRP concludes that the use of generic reference levels should *not* encourage a 'trade-off' of protective actions among the various components of the existing annual dose. In this regard the ICRP considers that a low level of existing annual dose does not necessarily imply that protective actions should not be applied to any of its components; and, conversely, a high level of existing annual dose does not necessarily require intervention. Should intervention be considered justifiable, the form, scale and duration of the protective actions should be optimised.

ICRP 82 generic reference levels in perspective

The identification of existing annual doses low enough to make intervention usually not to be expected, and not likely to be justifiable, is not simple and certainly not straightforward. For perspective purposes, it is helpful to use the 'natural' existing annual doses experienced in many parts of the world. The global average 'natural' dose is 2.4 mSv/a, but many large populations have lived for years in areas of the world experiencing typically elevated doses of up to around 10 mSv/a, with some populations even incurring doses above 100 mSv/a. In many of the places experiencing high levels of background radiation, the dominant component of exposure is that to the gas radon in dwellings; in other situations, the exposure is mainly caused by other gamma-emitting radionuclides, such as radium in soil and water.

With some exception, intervention has rarely, if ever, been undertaken to reduce the typically elevated 'natural' background doses of about 10 mSv/a. Moreover, only occasionally have protective actions been implemented to reduce higher 'natural' background doses, even when these doses were controllable. This might suggest that competent authorities have considered these levels as being unlikely to trigger any intervention in those situations.

Moreover, the ICRP considers that a high level of existing annual dose - *e.g.*, due to high natural background levels - should not justify *per se* a particular component of annual dose - *e.g.*, a high level of annual dose attributable to long-lived radioactive residues. This should always be restricted following the principles of the System of Radiological Protection for intervention. However, as the expected radiation health effects depend on the dose received and not on the source origin, the ICRP also considers that the typically elevated levels of existing annual doses from 'natural' sources, which have not triggered any protective action, may provide a useful insight into decisions related to intervention.

Further insight on sufficiently low levels of existing annual doses can be obtained from earlier recommendations given in ICRP Publication 63[2] and in ICRP Publication 65[3]. In these publications a number of intervention situations including some involving prolonged exposure were addressed. Specific reference levels below which any intervention or action is unlikely to be taken in various situations were here recommended, suggesting levels ranging from a few to a few tens of mSv for a dominant single component of the existing annual dose. Such intervention and action levels have been generally incorporated into international standards[4] and some national regulations. Again, this suggests that governmental authorities have considered the recommended levels (of around 10 mSv/a) as being unlikely to trigger intervention.

Response to nuclear or radiological emergencies - principles and experience

The International Commission on Radiological Protection (ICRP) has indicated that its basic framework for radiological protection is intended to prevent the occurrence of deterministic effects, by keeping doses below the relevant thresholds, and to ensure that all reasonable steps are taken to reduce the induction of stochastic effects. Although the ICRP policy for radiation protection has evolved over the years, its main objective has remained basically unchanged. It was formulated in the latest recommendations from ICRP (Publication 60) as: *The primary aim of radiological protection is to provide an appropriate standard of protection for man without unduly limiting the beneficial practices giving rise to radiation exposure*. The ICRP policy is *also* to supplement the available scientific knowledge by value judgements about the relative importance of different kinds of risk and about the balancing of risks and benefits.

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[2] INTERNATIONAL COMMISSION ON RADIOLOGICAL PROTECTION, *Principles for Intervention for Protection of the Public in a Radiological Emergency*. Publication 63, Pergamon Press, Oxford, New York, Seoul, Tokyo (1993).

[3] INTERNATIONAL COMMISSION ON RADIOLOGICAL PROTECTION, *Protection Against Radon-222 at Home and at Work*. Publication 65, Pergamon Press, Oxford, (1993).

[4] INTERNATIONAL ATOMIC ENERGY AGENCY, *International Basic Safety Standards for Protection Against Radiation and for the Safety of Radiation Sources.* Safety Series No. 115, Vienna (1996).

Protection principles in a nuclear or radiological emergency

The System of Radiological Protection makes a distinction between *source-related* protection - which is concerned with the exposures of individuals resulting from a single source - and *individual-related* protection - which is concerned with the exposure of a single individual from many sources. Provided that the individual doses are well below the threshold for deterministic effects, the contribution to an individual dose from a single source has an effect that is independent of the doses from other sources. For many purposes, each source, or group of sources, can then be treated on its own.

Source-related assessments make it possible to judge whether a *practice* or *intervention* is likely to bring benefits sufficient to outweigh any disadvantages and whether all reasonable steps have been taken to reduce the radiation exposures that a source will cause. They thus facilitate the justification of *practices* and *interventions* and the optimisation of protection at the source level. Source-related assessments take account of the magnitude (increase or decrease) of the doses attributable to the assessed source, and of the number of individuals exposed, but not of the influence on individuals of other exposure sources.

The system of radiological protection for intervention situations is based on the following general principles of *justification* and *optimisation*:

(a) *All possible efforts should be made to prevent deterministic effects.*

(b) *The intervention should be justified, in the sense that introduction of the protective measure should achieve more good than harm.*

(c) *The levels at which the intervention is introduced and at which it is later withdrawn should be optimised, so that the protective measure will produce a maximum net benefit.*

Dose limits used in the radiation protection system for practices do *not* apply in the case of intervention, for which intervention levels in terms of *avertable dose* should be applied.

Some reflections and lessons from past emergencies

Radiation emergencies in the past have demonstrated that immediately after the emergency phase of the response, there will be immense pressure from the public, public officials and the media to act to correct the problem and return the situation to normal. Without prior arrangements, public officials, when under intense pressure to restore the situation to normal, may take highly visible actions even if these are only minimally effective or even counterproductive. During the response to Chernobyl many unjustified efforts were carried out because of this pressure, such as decontamination of areas (*e.g.* Pripyat) that were evacuated and never resettled.

The Chernobyl and Goiânia accidents demonstrated that public officials make decisions concerning implementation of countermeasures affecting the public during the post-emergency phase of a radiation emergency. These officials were not radiation specialists and they made their decisions on the basis of their understanding of both the radiological risk and of societal and political concerns. This was recognized by the ICRP[1] when it recommended that guidance for taking post-emergency countermeasures based on scientific consideration of radiation protection should serve as an input into the wider decision-making process. It is important that the decision-maker understands the guidance for dealing with the radiological risk and be able to explain it to the public and the stakeholders for it to be useful as a decision-aiding tool. Assurance that the actions being taken will guarantee the "safety" of the affected populations should therefore be elaborated by the radiation protection community, disengaged from the linear non-threshold (LNT) risk hypothesis.

Following radiation emergencies the public took inappropriate and in some cases harmful action due to fear and misunderstanding concerning radiation risks and how to reduce them (*e.g.* refusing to buy products from the area, refusing to sell airline tickets to people from the area, having abortions due to a fear of radiation induced effects, and refusing to provide medical treatment to victims). These fears were in part due to the use of the LNT hypothesis by unofficial sources, the use of cryptic technical terms and the reluctance of technical experts to provide the definitive guidance needed and wanted by the pubic. Therefore, the LNT hypothesis should be reconsidered as basis for decisions on countermeasures.

Experience shows that international guidance does not address many post-emergency countermeasures that should be implemented, in part, based on radiation protection principles and insights. These include personal monitoring and decontamination, decontamination of property, release of contaminated property for use, initial medical screening, long-term medical follow-up, contaminated non-food products, and termination of countermeasures ('return to normality').

The ICRP[1] and others[5] have pointed out that it is impossible to anticipate or address factors not directly related to radiation protection principles when developing radiation protection guidance. Attempting to consider other factors or anticipate what would be acceptable to the public would only undermine the technical foundation of the recommendations, making them difficult to apply consistently, adjust or explain. It is the role of the radiation protection expert to give the best professional advice, even if the decision-maker, bowing to the pressure of political or public opinion, subsequently ignores it. Therefore, international guidance on interventions after an accident should be based *solely* on radiation protection considerations.

Applying ICRP 82 in an integrated framework of emergency response

A fully integrated system for implementation of countermeasures must have a sound technical basis but must also be understandable, explainable and acceptable to the public and to decision-makers. The essence of the guidance is that it must do more good than harm (be justified) and assure the public that they are safe. The ICRP 82 recommendations were developed for prolonged exposure situations, and they can be used as basis for an integrated system of radiological protection in emergency situations with special emphasis on the radiation protection of populations affected by nuclear accidents (*e.g.* Kyshtyhm, Chernobyl) or radiological accidents (*e.g.* Goiânia) in the post-emergency phase.

The radiation protection strategy for a population affected by a nuclear or radiological accident should first of all do everything possible to avoid serious deterministic effects and should thereafter implement protective actions with the aim of averting doses to the population to a *safe level*. Once all required protective actions have been undertaken, the situation should be considered 'normal' again with no further restriction being imposed.

Figure 2 illustrates the protection strategy after a nuclear or radiological accident: all effort should be done to avoid deterministic effects, and the reduction in expected stochastic effects should be based on optimized intervention and action levels to achieve a safe level, which can be defined as:

(1) From a radiation protection point of view "safe" means that population or critical subgroups will not receive a total annual dose leading to *identifiable*

[5] INTERNATIONAL ATOMIC ENERGY AGENCY, *Restoration of Environments with Radioactive Residues*. Proceedings of an International Symposium, Arlington, Virginia, USA, 29 November - 3 December 1999. IAEA, Vienna (2000).

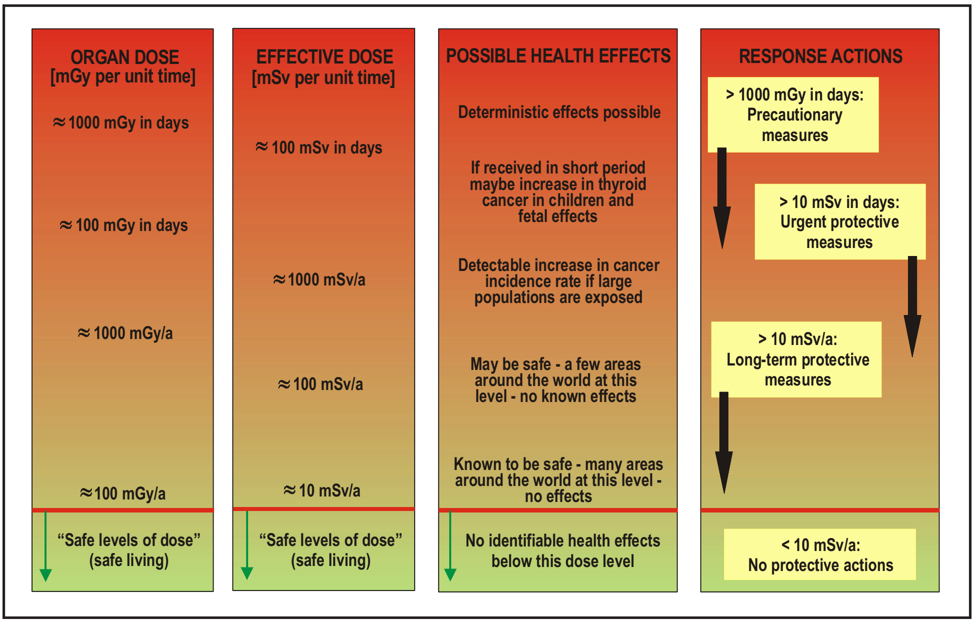

Figure 2. Protection strategy following a nuclear or radiological accident. At very high levels of individual dose preventive and urgent protective actions should be implemented to prevent deterministic effects and high probabilities of stochastic effects. At lower individual dose levels countermeasures should be implemented to avert doses to the affected population to reduce the stochastic effects and to restrict the residual individual dose level to a safe level. This level can be regarded as level for safe living conditions after a protective measure has been lifted (or considered but not implemented).

adverse health effects; "safe" does, however, *not* mean zero risk.

(2) Normal living conditions means that members of the public can live without any significant disrupting restrictions. Safe conditions are when people are living in normal living conditions or are following restrictions associated with radiation exposure.

(3) A total annual effective dose of about 10 mSv can be used as a reference level for safe living conditions. However, if a process of justification and optimization results in different (higher or lower) dose levels, these should be applied in that given situation.

The system of radiological protection for interventions can be rephrased based on the following principles:

(a) *Actions to avoid serious deterministic effects should almost always be undertaken*

(b) *Protective and remedial actions should be based on justified and optimised specific intervention levels and action levels*

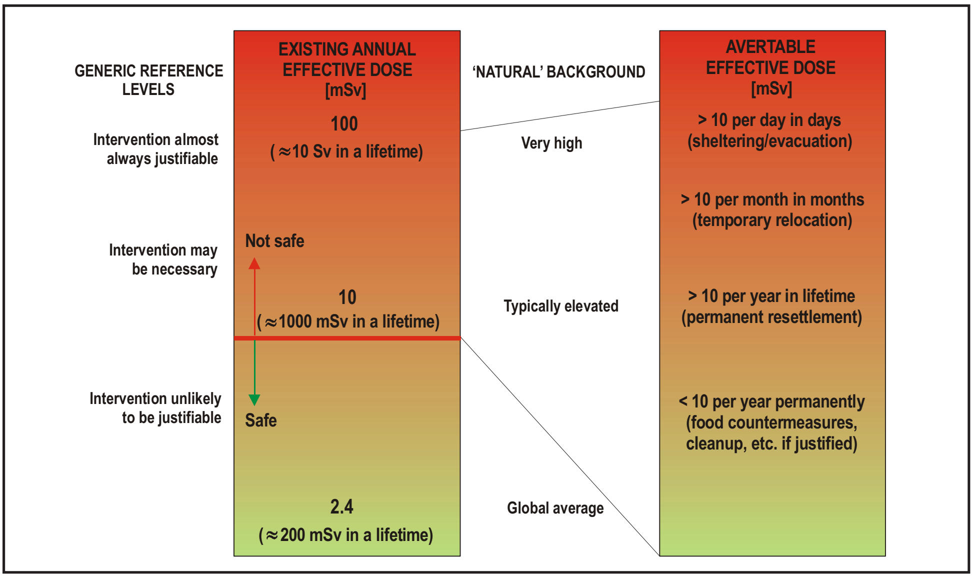

Figure 3. Schematic representation of the system of protection in emergency situations with an upper bound of the generic intervention levels of existing dose vis a vis the specific intervention levels of avertable dose. The specific intervention and action levels have been set to give an optimum reduction in individual doses. The exposure level to be compared with the intervention/action levels is the average individual dose in the critical group.

(c) *The population affected by a nuclear or radiological accident should be safe after protective and remedial actions have been implemented (individual effective doses < 10 mSv/a)*

The rephrased system of radiological protection for interventions in emergency situations is illustrated in Figure 3.

An annual effective dose of about 10 mSv (3 - 5 % additional average lifetime risk of fatal cancer if lifetime is taken to be 60 - 100 years) represents an upper bound on residual annual dose dividing exposure situations into two "classes". Situations with annual effective doses to the critical group above this level would normally not be considered as normal. In cases where the residual dose is characterized as "normal" it would henceforth be considered "background". In addition, a high level of existing annual dose should not preclude the introduction of a new practice, as a practice is controlled through the additional annual dose attributable to the practice rather than through the existing annual dose.

Conclusions

The Chernobyl and Goiânia accidents clearly demonstrated the need for recommendations on ‘normal’ or ‘safe’ living conditions in post-accident management. The USSR lacked criteria for implementation of countermeasures and return to normality (ending countermeasures) at the time of the Chernobyl emergency. In the years after the Chernobyl accident, the former Soviet Union - due to public pressure - adopted criteria for resettlement and other countermeasures that were not founded on established radiation protection principles. In the opinion of many radiation protection professionals, the criteria were not justified and probably have done more harm that good. During the response at Goiânia, it was very difficult to set operational levels for post-emergency intervention that were consistent with internationally accepted scientific principles because of time constraints and political pressure. This resulted in the use of the dose limit for practices as a basis for intervention and consequently in protective actions, generation of contaminated waste and decontamination and disposal costs that did not appear to be justified on radiation protection grounds.

The recommendations in ICRP Publication 82 were developed to fill a long experienced gap with regards to radiation protection against exposure from long-lived radionuclides in the environment, including those originating from radiological or nuclear accident. The recommendations are based on objective assessments of the health risks associated with prolonged exposure levels and on radiological protection attributes of various exposure situations. Typically elevated prolonged exposures due to natural radiation sources are usually ignored by society, while relatively minor prolonged exposures to artificial long-lived radioactive residues are a cause of concern and sometimes prompt actions that are unnecessary in a radiological protection sense. This reality of social and political attributes, unrelated to radiological protection, usually influences the final decision on the level of protection against prolonged exposure. Therefore, while ICRP 82 should be seen as a provider of decision-aiding recommendations mainly based on scientific considerations on radiological protection, the outcome of its advice will be expected to serve as input to a usually wider decision-making process.

Based on the radiation protection recommendations in ICRP 82 a dose level of *10 mSv per annum* can be used as a reference level for ‘safe living conditions’ or ‘return to normality’ after protective measures have been lifted (or considered but not implemented). The definition of ‘safe’ in the context of residual radiation exposure of population groups is that *no* radiation induced adverse health effects can be *observed*, and, equally important, that the residual risk for developing such health effects is low for the affected individuals.

Strålskyddsåtgärder när strålrisk föreligger

Anne Weltner
STUK

I direktiv VAL 1.1 beskrivs grunderna för beredskapsplanering med tanke på strålrisklägen, åtgärdsnivåer för igångsättande av centrala skyddsåtgärder, samt grunderna för strålskydd av dem som deltar i räddningsarbetet.

Denna anvisning är avsedd att användas av räddningsmyndigheter som ansvarar för beredskapsplanering, ledning av räddningsverksamhet och varningar till befolkningen samt av andra myndigheter som deltar i handhavandet av ett läge med strålrisker.

Direktiv VAL 1.1 finns på internet publicerat av STUK och Finlex:
på svenska:
i html-format http://www.stuk.fi/saannosto/VAL1-1r.html
i pdf-format http://www.finlex.fi/pdf/normit/6430-VAL1-1r.pdf

på finska:
i html-format http://www.stuk.fi/saannosto/VAL1-1.html
i pdf-format http://www.finlex.fi/pdf/normit/6429-VAL1-1.pdf

Innehållsförteckning

Development of a prototype radiation surveillance equipment for a mid-sized unmanned aerial vehicle

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Abstract

A prototype radiation surveillance equipment has been developed to be used in a mid-sized Ranger unmanned aerial vehicle (UAV) acquired by the Finnish Defence Forces. A multi-detector assembly was designed for the acquisition of dose rate and radionuclide concentration in the release plume. Detector assembly includes a GM-tube based dose rate meter, an inorganic scintillator detector and a semiconductor detector operating at room temperature. A sampling unit was designed for the collection of an aerosol sample of the plume for a detailed analysis in a ground based laboratory. The measurement data from all three detectors and several environmental parameters are collected by the onboard data acquisition computer. Real-time data dissemination is implemented with a TETRA based radio network.

Test flights have been carried out with target drones and a small manned airplane. The Northrop KD2R-5 target drones have been used to simulate the high-G launch and vibration environment of the Ranger aerial vehicle. Target drones have been used because their air vehicle classification allows small test packages to be installed without tedious air safety protocols. Stability and survivability of the detectors, GPS navigation and radio frequency communication have been studied with the target drone test flights. Ground station software was developed to visualise the measurement data and to track the position of the air vehicle on a digital map. Test flights with the small manned airplane have been used to study the operational aspects of the detectors with greater detail.

The housing for the instruments has been designed and constructed based on the experiences gained with the test flights and the laboratory

measurements. The housing satisfies the aviation authority standards. Special attention has been paid to the high modularity, quick installation and ease of use.

Introduction

Finnish Defence Forces decided to acquire the Ranger unmanned aerial surveillance system at the beginning of the year 1999. Soon after a feasibility study for the use of the Ranger system in radiation surveillance missions was started at STUK. The main motivation was to reduce the radiation risks imposed on the crew of conventional manned surveillance flights. Unmanned aerial vehicles provide a new platform for in situ measurements inside a nuclear accident release plume. Ranger system was found to be suitable for such missions and the development of a radiation surveillance equipment for the Ranger system was initiated.

Unmanned aerial vehicle platform

The Ranger unmanned aerial vehicle (UAV) is mainly designed for electro-optical surveillance in military missions, but the modularity of the payload enables it to carry out a wide range of missions. Ranger UAV has endurance of five hours and it can carry 40 kg of payload. The range of the command and data link is approximately 150 km, but the air vehicle is capable of performing autonomous operations without the command and data link. This property gives the maximum range of 500 km if the cruise speed of 200 km / h is used. With these characteristics, Ranger UAV can be used for radiation surveillance missions as well.

Range and duration of the flight of the Ranger enables it to cover large enough area needed for locating and mapping of a radioactive plume. The payload capacity of the Ranger enables it to carry reasonably large detectors, but rules out the use of HPGe detectors due the volume and weight requirements of the cooling system.

Prototype radiation surveillance equipment

The limitations of the platform led to the selection of a 6" x 4" thallium activated sodium iodine scintillator detector as the main detector. This kind of detector provides fairly good sensitivity and enables the detection of the plume from a distance. The poor energy resolution of the NaI(Tl) detectors hinders the use of the main detector for nuclide identification.

A cadmium zinc telluride (CZT) detector was chosen as a secondary detector to compensate the low energy resolution of the main detector. CZT is a semiconductor and the detector has a fairly good energy resolution. Secondary detector increases the usability of the system for nuclide identification. The CZT detector is not cooled and is operated in ambient temperatures. This limits the thickness of the detector to the millimeter scale. The small size and poor efficiency above 100 keV limits the use of the detector to in situ measurements in a high radiation field. A filter sampler equipped with an air flow meter has been installed to further improve the measurements. Filter samples can be analysed post flight in a ground based laboratory for nuclide concentration estimates in the plume.

A Geiger – Müller tube based dose rate meter completes the detector assembly. A commercial dose rate meter has been modified to provide single pulse counting for short integration times. The short integration time improves the spatial resolution of the dose rate measurement on a fast moving platform.

The onboard data acquisition computer provides data storage and detector electronics control capabilities for the system. The computer is based on PC/104 standard modules and has a Pentium class processor that runs with Windows Embedded operating system. Data is stored on a silicon chip based Disk-on-Chip drive. The computer also provides A/D converters for environmental sensors, counters for GM tube pulse counting and GPS receiver for the spatial positioning of the measurements.

Real time data transfer with TETRA radio network

Real time data transfer is needed for the best possible usability of the system. Some radiation surveillance missions require the UAV to fly autonomously outside the data link coverage.

A secondary data link is needed for the real time data transfer.

A nation wide TETRA based radio network called VIRVE is currently being built in Finland.

The VIRVE network is limited to official use and can provide adequate resources in emergency situations. All measurement data (spectra, pulse counts, environmental parameters etc) will be transmitted directly to the STUK headquarters for further analysis using TCP/IP network provided by the VIRVE radio network.

Flight tests with target drones

System components have been tested in Finnish Defence Forces Northrop KD2R-5 target drones. Target drones have been used to simulate the high-G launch and vibration environment of a UAV. Various components have been tested and found to be durable enough to withstand the demanding conditions. Detector stability has been studied as well. Scintillator detector has been proven to be stable in the vibrational environment of a target drone and consequently will be stable also in the Ranger UAV. Variation in the temperature of the detector has been shown to be the main cause of instability. Variations can be attenuated with adequate insulation and heating of the detector housing.

Test flights have been used to study the reliability of GPS tracking and radio frequency data communications. Various data acquisition schemes have been programmed and studied together with data visualisation and geocoded information concepts. Commercial 'off the shelve' GPS receiver was found to be reliable in fast moving airborne platforms. Radio frequency data transfer with commercial transmitters at UAV speeds has been proved to be reliable enough to provide the bandwidth needed.

Figure 1. The CAD design of the instrument housing

Equipment housing design

Equipment housing has been designed to meet the aviation authority standards. Composite materials have been used to construct light weight and durable housing that can withstand the foreseeable situations. The payload space of the Ranger air vehicle provides multiple equipment bays, but compact design which places all equipment at a single equipment bay was selected to emphasise the fast and easy installation. The new housing constructed will replace the standard camera payload of the Ranger.

Conclusion

A multiple detectors system has been constructed for the RANGER unmanned aerial vehicle. The system and the real time data transfer has been tested in the laboratory and with the help of several test flights in target drones and in conventional aircrafts. Performance of the complete system in the Ranger will be tested in the near future.

Radiological emergency preparedness training at Loviisa NPP

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Introduction

The consequences of nuclear power events or accidents can be effectively mitigated by countermeasures conducted by emergency organization. The protective actions recommended to the public and radiation protection of the personnel working at the plant have a high priority. The protective actions are based on radiological assessment. By real-time monitoring and theoretical calculations the emergency organization aims to assess and predict the accident sequence: plant process, core damage, transportation of radioactivity from the fuel to the reactor coolant, containment, plant buildings, stack and to the environment.

The competence of the emergency organization is based on each individual's capability for the normal time job position. The specific emergency training consists of topical training for each branch and real-time co-operation exercises in which management and various sections participate together. This presentation concentrates on the radiological part of emergency preparedness training at Loviisa power plant.

Emergency preparedness organizations

Licensee emergency management organization

The shift personnel 24 h/d at the plant consists of 15 men operating the two units and three men at fire brigade and the security personnel. The safety engineer on duty can stay outside the plant but must always to be able reach the plant within 40 minutes after a call. If the shift supervisor declares site emergency or general emergency Fortum activates its emergency organization. As a lower alternative he can declare alert which means that 10-15 persons from the emergency organization are called to enter the power plant.

The plant manager, or his substitute, shall direct the emergency organization. The plant emergency management team settle themselves at the

Emergency Operations Facility (EOF), which is located in a civil defense shelter below the plant administrative building. The emergency organization consists in addition to the plant personnel also engineering support provided by Fortum Nuclear Services (FNS) and Fortum communication personnel which are located in Helsinki.

FNS engineering support has a broad technical expertise concerning nuclear accidents - consequences and counteractions. It has also radiological experts with experience on radiological accident analysis, PSA study, release estimates, radioactivity / dose rate –calculations and transportation of radioactivity & doses in the environment.

Off site emergency organizations

Numerous authorities at local, provincial and national level contribute to off site protection activities. The fire chief of town Loviisa has the responsibility of rescue service management. When the accident has larger dimensions, the provincial board or the ministry of interior can appoint a higher level rescue service manager and then the local fire chief concentrates on the rescue service management on the county level. He supervises a rescue service management group that gathers in the County Emergency Operations Center (CEOC) at the town of Loviisa, 15 km from the power plant.

Radiation related topical training

Radiation engineers

Radiological assessment and radiation monitoring are key areas in the emergency organization of a nuclear power plant. The aims are to make assessment of the situation and predict the accident sequence: plant process, core damage and transportation of radioactivity from the fuel to the reactor coolant, the containment, plant buildings, stack and to the environment. Based on these predictions the emergency management can direct plant personnel and people in the vicinity.

Trained areas are radiological assessment, emergency organization management, information, radiological surveys, taking and analyzing samples from the reactor coolant and containment atmosphere. Emergency preparedness training of radiation engineers consists of training on these topics and real-time exercises together with the rest of emergency organization.

Most of radiation engineers have during normal plant operation radiation related responsibilities and therefore they have a fair good base for emergency preparedness training. Special issues included in emergency preparedness training are emergency plan and implementing procedures, use of plant process computer displays and radiological calculations.

Environmental calculations are calculated by LENA (LENA is an acronym from “Liten, Enkel, Numera Användbar”). It has been developed by the Swedish authority SSI and used also by Swedish nuclear power plants. A new area is a support system for radiation specialists SaTu (SäteilyAsiantuntijan Tuki-järjestelmä – support system for radiological experts). It is an excel-based computer code that predicts the radiological consequences based on plant process data. Because nuclear power plant accidents are complicated so also the use of SaTu is a demanding task.

In larger co-operation exercises the radiation engineers also practice supervising health physics personnel, communication with other emergency organization and contributing to information to mass media.

Health Physics

Health physics personnel has long experience in radiation protection at Loviisa nuclear power plant. During annual maintenance outages they monitor and potential / actual contamination radiation within the plant and supervise the appropriate radiation protection actions. In case of an accident the radiation levels may be many orders of magnitude higher.

Radiation chemistry

The regular training includes taking and analyzing samples from the reactor coolant and containment atmosphere make use of accident sampling points and methods.

Rescue service authorities

STUK - Radiation and Nuclear Safety Authority of Finland represents the Finnish safety authority that sets the regulations for the use of radiation and nuclear energy and supervises that they are followed. STUK also co-operates in rescue service actions. STUK observes the annual emergency exercises at Loviisa power plant and participates with its own emergency organization in this exercise. The fire brigade of town Loviisa drills regularly with the plant fire brigade.

At least every 3rd year there will be a larger rescue service exercise with national and regional authorities. These exercises require thorough advance planning and training. According to YVL 7 .4 the licensee also contributes to the topical training of rescue service authorities for nuclear accidents.

Co-operation exercises with real-time simulated exercises

Licensee level emergency exercises

The activities in licensee level emergency exercises are:

- management of and co-operation between various organizational units,
- alerting and activating the emergency organization,


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• investigating the plant situation and assessing the progress of the accident,
• control room operations,
• working procedures in emergency operations facility (EOF)
• extinguishing a fire,
• gathering and evacuating the personnel,
• radiological surveys,
• taking and analyzing samples from reactor coolant and containment atmosphere
• estimating and forecasting fuel damage, radioactive releases and radiation doses,
• preparing recommendations concerning protective actions.

In real time exercises the emergency organization comprises of the power plant management, health physics, radiochemical laboratory and control room operators. The emergency exercises encompass also task groups, information personnel, the plant fire brigade and the engineering support at Helsinki.

STUK and the management of the local rescue service authorities (fire brigade, police, health center, information bodies etc.) have also participated in these emergency exercises.

Accident simulation in emergency exercises

In emergency exercises, in which the "backbone" of the emergency organization participates, the accident descriptions comprise of three semi-independent entities: the plant process sequence, the radiological situation at the plant and the consequences of the radioactive release to the environment.

The on site radiological description defines the quantity and timing of the radioactive releases to the reactor coolant, to the containment and other plant areas and into the environment. It presents also the postulated radiation levels at the plant.

The environmental description includes radioactive release to the environment, weather data, the atmospheric dispersion and deposition and the external dose rates and dose commitments.

Conclusion

Among the public there are different point of views towards nuclear power. However, people seem to be rather unanimous in one sense: exercising for nuclear accidents is needed.

Emergency response planning around Norwegian research reactors

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Introduction

The Norwegian Radiation Protection Authority (NRPA) has for some time focused on emergency response planning in connection with Norway's two research reactors. This refers to emergency response planning off site, where the local police, health authority, municipal authority, county governor and central authorities headed by the Crisis Committee for Nuclear Accidents are key players in dealing with an accident. The NRPA is concerned not because planning at the facilities, by the police or by the central authorities is inadequate, but because planning is insufficiently coordinated.

In order to make all parties involved in handling an accident situation aware of their own and others' responsibilities, the NRPA, in keeping with international recommendations, has worked to achieve a basis for harmonising planning among the various players. The International Atomic Energy Agency (IAEA) has published a document in its Safety Standards Series entitled "Preparedness and Response for a Nuclear or Radiological Emergency". This standard underlies our effort to achieve coordination of emergency response planning around Norway's two research reactors. Both are run by the Institute for Energy Technology (IFE) and are located at Halden (25 MW) and Kjeller (2 MW) respectively.

Background

As a background to the issues we have discussed in this process, some important points in the IAEA standard are reviewed:

- The standard calls for emergency planning and response to be based on a safety analysis, and for a broad range of postulated events to be included in the threat assessment. Scenarios involving a combination of a radiological emergency and a conventional emergency (e.g. earthquake) should be included.
- Various threat categories are defined. Nuclear power stations are classified in threat category I, whereas most research reactors are classified in

threat category II. Further facilities and sources are listed, ending with activities which do not usually involve radiation sources but which can become contaminated as a result of events in category I and II.

- A system for classifying types of emergencies is defined. For research reactors these are defined as follows:

• *"General emergency"* (actual release or risk of release warranting taking of urgent protective actions off the site)

• *"Site area emergency"* (major decrease in the level of protection for those on the site and off the site which requires planning for action to be taken off the site)

• *"Facility emergency"* (major reduction in level of protection for personnel on the site, but no threat off site)

• *"Alert" (uncertain or significant reduction in level of protection for the public or personnel on the site)*

• *"Uncontrolled source emergency"* (loss or theft of a radiation source).

ENATOM, the IAEA's Emergency Notification Manual compiled in conformity with the Early Notification Convention, also refers to *transboundary emergency*.

- Various planning zones are defined for the various threat categories. "Urgent protective action planning zone" is relevant for research reactors. In this zone arrangements should be made for urgent protective action to be taken promptly to avert dose off the site in compliance with international standards.

Organisation and legal basis

The role of the radiation protection authorities in the Nordic countries differs both as regards emergency response planning and accident management. In Norway the NRPA is the competent authority in this field. The Crisis Committee for Nuclear Accidents, headed by the Director General of the NRPA, is for responsible for action taken at an early stage of an accident occurring in Norway or elsewhere. Minor accidents are handled by the NRPA. The police at the accident site will be responsible for saving lives and protecting health in cooperation with the fire service and the health service/ambulance service. In the event of major accidents the local police will be a resource available to the Crisis Committee and will be responsible for implementing a number of dose-reducing measures as soon the Crisis Committee becomes operational.

The Act on Radiation Protection and Use of Radiation (of 12 May 2000) and Royal Decree of 26 June 1998, require the IFE and the participants in the emergency response organisation at the central, regional and local level to adhere to a coordinated body of plans. The Radiation Protection Act, the Nuclear Energy Act and the Health Preparedness Act require the various accident management players, and anyone likely to be affected by an accident, to maintain emergency response programmes. The legislation also imposes information and notification duties on the IFE and the nuclear emergency response organisation.

Issues

Our work on developing a basis for coordinated emergency response plans for crisis management in connection with accidents at Norwegian research reactors has greatly benefited from IAEA standards and other documents. However, various issues needed clarification. Some of these are presented below.

What scenarios should planning be based on?

According to the IAEA standard, a broad range of scenarios should be included. The IFE's safety report, which includes a "reference accident", is a natural starting point. Since 11 September last year a variety of terror scenarios have been in focus, but such scenarios are not covered in the safety report. Despite this, we have chosen to use the report as a basis. A flexible body of plans is called for which gives greatest emphasis to the most probable scenarios but which is also suited to other scenarios and other types of accidents. This justifies exercises involving scenarios of greater severity than the reference accident described in the safety report. In the aftermath of 11 September we have also discussed physical protection and the establishment of a *"design basis threat"*, i.e. the attributes and characteristics of a person or organisation who might attempt unauthorized removal of nuclear material or sabotage which may have safety or health consequences for employees, population or environment. We aim to integrate thinking on physical protection and preparedness more closely into the overall approach, and are working on ways of achieving this.

Classification of accidents at national facilities

In the standard IAEA proposes 5 degrees of accident severity, both for nuclear reactors and for research reactors, in addition to *"transboundary emergency"* in ENATOM. In other words, there are many classes to keep in mind and plan for, at the same time as the classes described do not in our view necessarily describe the degree of severity faced. We note, for example, that an *"alert"* can be just as serious as a *"facility emergency"*. For the time being we aim to employ just three of the classes mentioned: *"general emergency", "site emergency"* and *"alert"*.

The Crisis Committee is assigned a role both in connection with accidents in Norway and with accidents and events elsewhere. The nuclear emergency response organisation has two levels of preparedness:

- *Information preparedness*, which is declared in situations requiring the nuclear emergency response organisation to be informed, and
- *Heightened nuclear emergency preparedness*, which is declared in situations which may have consequences for Norway or Norwegian interests.

These levels of preparedness are thoroughly incorporated into the nuclear emergency response organisation and are also described in the media when appropriate (e.g. the Kursk accident, where a state of information preparedness was declared). It is important to be able to employ these levels in the event of a national accident too. We therefore see a need to develop a “translation” between accident classification and preparedness levels. We also consider the merits of coordinating preparedness levels, accident classification and INES classification, or whether to make less use of the INES scale when informing the population in an accident situation.

Having preparedness levels as well as a system for classifying national events could lead to confusion. The preparedness levels will be used both internally and externally, but, especially in the local arena, publicising the classification of an accident will be significantly more informative.

Changes in and /or introduction of new planning zones at existing facilities

A 300 metre building prohibition zone is currently enforced around the Kjeller reactor. This zone serves no purpose in relation to preparedness planning, and the police have established their own evacuation zones. We see benefits of harmonising these zones as far as possible, not least with a view to communication.

Appropriate information to the population

Neighbours of the IFE facility ask for information on preparedness plans and on what they themselves can do in an accident situation. In the event of an accident the police, in line with their emergency plan, will distribute an information card to the public explaining what action to take. We see a need for more information to the population in the event of an accident. This needs to be developed separately.

Those likely to be affected should be involved at an early stage in this ongoing process.

Conclusion and further work

Through our work on establishing a basis for harmonisation of emergency preparedness plans in relation to potential accidents at the IFE's research reactor, our aim is to contribute to a clear-cut division of responsibilities, professionally well-founded measures at the acute stage of an emergency, predictable emergency management, simplified information flows and intuitive awareness among the parties and, not least, coordinated information to the population and others who may be affected. The NRPA is accordingly preparing a guide in conjunction with the IFE, and in due course with the other actors, with a view to clarifying responsibilities and planning assumptions. This guide will be binding on many actors by virtue of the Radiation Protection Act or other legislation. What remains to be done at the national level is to involve the police, municipal authorities, county governors, the health service, the Crisis Committee and the population and industry in the area surrounding the reactors.

As a small country with limited nuclear activity, we are concerned to ensure that the thinking on various aspects of safety, physical protection, information and emergency preparedness forms a coherent whole. We see inconsistencies in the basis established for systems for physical protection, safety at facilities and emergency preparedness planning. Perhaps "design basis accident" (safety) and "design basis threat" (physical protection) need to be accompanied by better guidance when it comes to establishing a "design basis for emergency planning and response".

Here in Norway we would like to see the IAEA's standards form the basis for a review of emergency planning in connection with port calls by nuclear powered vessels, along with an extension of the methodology to embrace radiation sources. This would promote uniform management of various events, geared to the degree of severity of the event in question.

However, we see that this work has an international dimension. There are clear-cut advantages to achieving the maximum possible international harmonisation of emergency response planning for various facilities. This will simplify information flows and other countries' chances of intuitively comprehending the situation and the measures implemented. We have therefore opted to follow the IAEA's recommendations as far as possible in our effort to achieve a basis for harmonising emergency planning and a coordinated response. Even so national adjustments and simplifications have proved necessary, and for that reason we diverge somewhat from the IAEA standard. In an international perspective this is detrimental, and we see advantages to international cooperation to reduce disparities in planning. We would also like to see greater harmonisation of emergency planning among the Nordic countries.

Posters

5a Finnish early warning system for nuclear emergencies: Experiences during 2000–2001

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Introduction

The Finnish early warning system for nuclear emergencies consists of the nation-wide automatic external dose-rate monitoring network and the control system, which is known as USVA. The monitoring network comprises of approx. 300 AAM95 central stations and substations equipped with RD-02 or RD-02L GM tubes. Stations are thoroughly inspected on a routine basis every 4–5 years. During the inspection the GM tube, battery and power adapters of the station are replaced with new ones, as is also the modem if necessary, and the alarm connections to USVA are tested.

The USVA system utilizes advanced www technology (including a browser-based, easy-to-use user interface) and a network of PCs with dedicated tasks. USVA's central hardware is located at STUK. USVA began its operation in the beginning of the year 2000, replacing the older y2k incompatible system. In a routine situation, USVA collects the monitoring data once a day. The USVA is capable of connecting (via telephone lines) to eight AAM stations concurrently, and the results from the whole country are obtained in about 15 minutes. When the system receives an alarm message from the network, it sends a text message to the mobile phones defined in a separate list and starts an automatic data collection procedure at all the stations situated within a certain distance (100 km) from the alarm-causing station.

Experiences

In 1999 efforts were to a large extent concentrated on building the new control system, USVA. Consequently, the operational state of the monitoring network degraded. Therefore, in the spring 2000 resources were channelled to the maintenance of the network. The effect of this resource input can be clearly

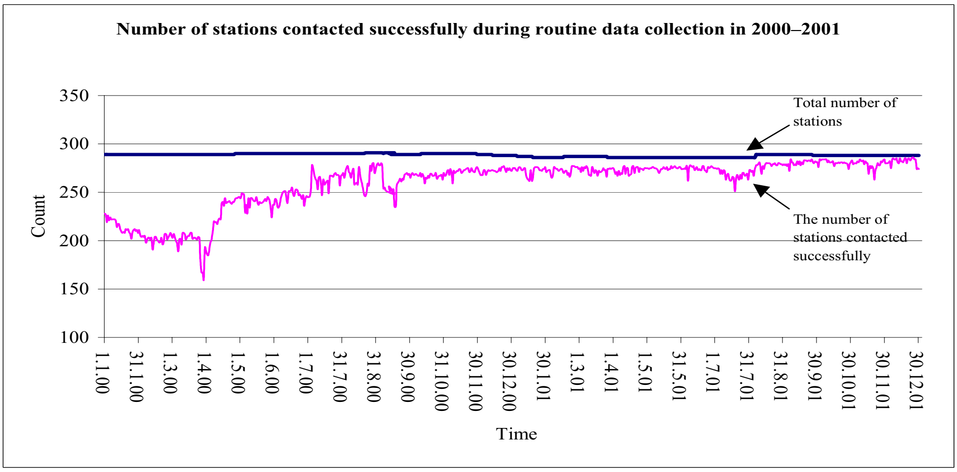

Figure 1. Number of stations contacted successfully during daily routine data collection in the years 2000–2001.

seen in Fig. 1: the number of the stations contacted successfully during daily routine data collection began to increase remarkably during the summer 2000 and has now reached an average value of 275 stations (corresponding to the percentage of 96 %).

In the year 2000 the monitoring network generated 26 automatic messages (see Table 1), most of those being system-identified hardware failures caused by problems with the backup battery (16) or with some other piece of equipment (3). The number of alarm messages was seven; four of these were false alarms caused by broken GM-tubes while three were real dose rate alarms. During the last year the total number of messages was 18. Backup battery failures caused three messages, as did also other system-identified equipment errors. One of the alarm messages (12) was a real one but the rest were generated by miscellaneous hardware errors.

Two of the four above-mentioned real dose rate alarms were due to careless use of portable dose-rate-meter calibration sources close to the GM tube of the automatic monitoring station. One alarm was caused by an industrial X-ray device, which was used to inspect welding seams of direct heating pipeworks near the monitoring station. The fourth alarm was caused by a fire fighter who, after having had radiological treatment in the hospital, returned to the fire station and walked too close to a monitoring probe.

As regards the USVA system, it has been functioning very well. A few errors have been encountered in the www server system but they have not affected the capability of USVA to retrieve data from the monitoring stations or

Table 1. Messages from automatic monitoring network during 2000–2001.

Year	Total number of messages	Hardware failures automatically identified as such by the system	Alarm messages[a] False	Alarm messages[a] Real
2000	26	19	4	3
2001	18	6	11	1

[a] Lowest alarm limit is 0.4 μSv h^{-1}

to handle alarms. Because USVA has been designed and realized mainly by STUK's experts, all necessary modifications and error repairs can be carried out in a fast, flexible and reliable way.

Summary

As compared to the late 90's, the reliability and operational quality of the Finnish early warning system has improved considerably during the last two years. This is due to the proper and well-planned maintenance of the dose rate measuring network and the introduction of the new control system, USVA, in the end of 1999. It can be said without hesitation that USVA has exceeded all expectations set on it.

5b Nuclear threats in the vicinity of the Nordic countries – a database, Nordic Nuclear Safety Research

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Introduction

The acute phase of an accident and the possibility of high exposure of the populations are always the most important threats in the emergency preparedness work. Radioactive contamination from an accident can however also cause long time effects for land use and enhanced doses to special population groups and economic problems for agriculture, grazing animals, reindeer industry, hunting, freshwater fishing, tourism and recreation. For planning purposes it is always valuable to be aware of potential radiation hazard and other potential threats in the vicinity of the Nordic countries. Thus, mapping such threats in a Nordic context is an important factor in emergency preparedness in the Nordic countries.

The project has dealt with threats from the north west of Russia and the Baltic states. The results from the different activities in the project is generated in a web based database called the "the base of knowledge".

The "base of knowledge" on nuclear threats

This project was one of the new cross-disciplinary studies in the NKS research program 1998-2001. The main task for the project was to aggregate already conducted knowledge of nuclear threats in the vicinity of the Nordic countries, into a "base of knowledge", present by modern information technology. This web based "base of knowledge" will be available for the Nordic authorities as a supplement for the national emergency preparedness work. This base of knowledge will, by information technology, be made available to authorities, media and the population. The users of the websites can easily get information on different types of nuclear installations and threats.

The first stage of the project was to prepare a list of relevant papers and reports that have been produced concerning nuclear threats in the vicinity of the Nordic countries, a literature database. The literature database was created as a part of the "base of knowledge" and is a database with the most relevant publications, papers and reports that have been produced regarding possible nuclear threats in the vicinity of the Nordic countries. The literature database is presented on a website and as a report with 500 references. As a summary of the literature in the database there are made two status reports on the most important issues of the project, threats from the nuclear power plants and the nuclear vessels. The reports give an overview of the work done in this matter. The reports are published as NKS reports.

At the Workshop 2000 experts from the different Nordic countries presented each countries evaluation of the threats against their country. There were presentations from the different Nordic countries concerning the threats from nuclear installations and there were discussions about source terms, models and consequences of nuclear threats.

The results of the discussions at the workshop and the presenting literature reports on the threats from nuclear power plants, nuclear powers ships and storage and handling of used fuel and radioactive waste are presented in the "base of knowledge".

SESSION 6. NON-IONIZING RADIATION

Non-Ionizing Radiation - sources, exposure and health effects

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Introduction

Non-ionizing radiation contains the electromagnetic wavelengths from ultraviolet (UV) radiation to static electric and magnetic fields. Optical radiation consists of UV, visible and infrared (IR) radiation while EM fields include static, extremely low (ELF), low frequency (LF) and radiofrequency (RF) fields. The principal scientific organization on non-ionizing radiation is the International Commission on Non-Ionizing Radiation Protection (ICNIRP). The main activity of ICNIRP is to provide guidance on safe exposure and protection of workers and members of the public by issuing statements and recommendations.

Ultraviolet radiation

The sun is the most significant source of natural UV exposure, whereas welding arcs are the most important artificial sources of UV and bright light in terms of health hazards. Usually welding is performed manually at construction sites, shipyards and in small workshops, so that in addition to welders, other workers close to the welding site are also exposed.

Erythema (sunburn) is an acute injury following excessive exposure to UV radiation. Erythemal threshold doses vary with skin type and colour (Table 1).

Though painful erythema will fade after a few days, there may still be serious late effects, e.g. skin cancer. The three common forms of skin cancer are basal cell carcinoma (BCC), squamous cell carcinoma (SCC) and malignant melanoma (MM). Around 90% of skin cancer cases are of the non-melanoma cancers, BCCs being approximately four times as common as SCCs. Exposure to UVR is considered to be a major etiological factor for all three forms of cancer.

Exposure to UV also causes a variety of eye disorders to the cornea, lens and retina. Cataracts are the most frequent effect, while photokeratitis

Table 1. Classification of skin types: susceptibility to sunburn and ability to tan.

Skin phototype	Sun sensitivity	Sunburn susceptibility	Tanning ability	Classes of individuals
I	Very sensitive	Always sunburn	No tan	Melano-compromized
II	Moderately sensitive	High	Light tan	Melano-compromized
III	Moderately insensitive	Moderate	Medium tan	Melano-competent
IV	Insensitive	Low	Dark tan	Melano-competent
V	Insensitive	Very low	Natural brown skin	Melano-protected
VI	Insensitive	Extremely low	Natural black skin	Melano-protected

(snowblindness), keratopathy and pterygium (growth on the conjunctiva) also result from UV exposures.

Since not all wavelengths of UV radiation equally effective in producing a biological effect, an action spectrum defines the relative effectiveness of different wavelengths. This relative response curve is normalised to a maximal value of 1.0 at the wavelength of maximal tissue sensitivity.

Visible radiation

As for the adverse effects of visible radiation on the eye, chronic over-exposure to short-wave light, so called blue-light, is a contributing factor to degeneration of the retina. Though the eyelids close at exposure to bright light, this response is too slow in special situations. Especially hazardous are laser beams focused to a small spot at the retina. Also exposure to bright light from welding arcs or metal halide lamps may cause acute or chronic ocular damage.

Electromagnetic fields

People in technically developed countries are surrounded by low and high frequency EMFs. Exposure to EMFs fields induces body currents and energy absorption in tissues. Thermal effects caused by temperature rise are obvious, whereas the mechanisms for suspected non-thermal effects are still unclear.

The main sources of occupational exposure to EMFs are induction heaters, RF welding equipment, electronic article surveillance devices, and medical and therapeutical devices. For the general public, the primary sources are high voltage powerlines, radio and television transmitters, and electrical appliances at home. Recently, as a result of the expansion of the use of hand-held cellular phones,

Table 2. Technical properties of six wireless telephones.

System	NMT 450	GSM 900	DCS 1800	CT1	DECT
Type	Analogue	Digital	Digital	Analogue	Digital
Pulse repetition rate	-	217 Hz	217 Hz	-	100 Hz
Peak power	15 W	2 W	1 W	10 mW	250 mW
Mean power	0.15-15 W	2-250 mW	1-125 mW	10 mW	10 mW
Cell radius	2-50 km	0.1-35 km	0.1-35 km	500 m	300 m

health concerns of people have focused to mobile phones and base station antennas. Table 2 indicates some technical properties of commonly used mobile and cordless telephones.

New varieties of cellular phones are under construction all the time, such as the Universal Mobile Telecommunication Systems (UMTS). The spectra of UMTS and some existing networks are shown in Fig. 1.

Base stations for mobile phone networks transmit and receive signals to and from a mobile phone. The power density generated by a base station antenna depends on the distance and direction from the antenna. The antennas are normally located so that the RF energy is directed away from the building to cover the surrounding area. In the direction of the main beam there is a free

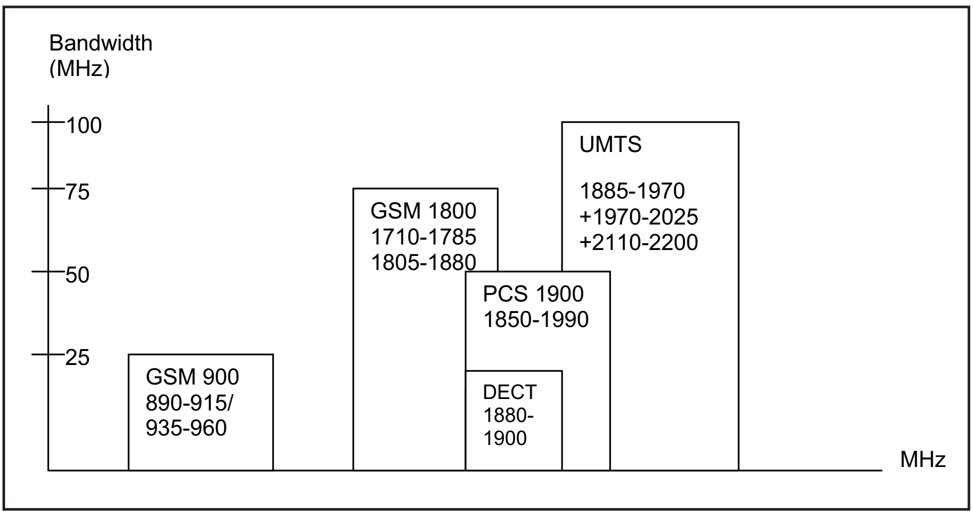

Figure 1. Spectra of various wireless networks.

Table 3. Maximum and average power densities measured at 26 sites in Sweden.

Location	**Max (mW/m^2)**	**Mean (mW/m^2)**
City	3.01	0.8
Town	0.049	0.034
Rural	0.006	0.0016
Indoors	0.0115	0.0055
Office (1.6 m to DECT base station)	3.7	0.154

space of typically several hundreds of meters to ensure the proper functioning of the mobile network. According to the field measurements taken at 26 sites in Sweden, most of the environmental EMF exposure was emitted by GSM 900 antennas (47 %). Television transmitters had a mean contribution of 48 % in rural areas and 13 % in the city. Measured maximum and mean power densities in different sites are given in Table 3. The levels of these fields are so low that temperature rise is unlikely, but other adverse effects have been suspected, such as EM hypersensitivity and development of cancer.

Relevant literature

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Perspective on electromagnetic fields as a potential genotoxic risk – where are we today?

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Introduction

Several thousand scientific articles published over the past twenty years relate to the question of whether electromagnetic fields, EMF, somehow can be involved in the induction or promotion of genotoxic effects including cancer. In a smaller number of these articles it has been stated that EMF may be the cause of such dangers. It is therefore important to review the present knowledge and also to clarify certain scientific aspects.

To the general public, the word "radiation" in many cases is wrongly perceived as threatening. It is there-fore important to note that ionizing radiation such as X-rays and gamma radiation gives rise to completely different physical and biological effects and carry a much higher energy per "photon" than EMF.

What follows is thus a summary of an analysis according to the so-called Hill's criteria, a review of some of the previous research that has been made for low frequency EMF (50/60 Hz) including some theoretical and experimental aspects of EMF. Finally a remark is made regarding "possible risks".

Reviews and Hill's criteria for cause-effect relationships

At least fifteen national or international committees composed of distinguished and competent persons and set up by responsible public bodies and professional associations have arrived at the conclusion that low frequency EMF does not constitute a cancer risk.

The criteria for the establishment of a solid cause-effect situation, as specified by A. Hill[1] are:

[1] A.B. Hill, The Environment and Diseases: Association and Causation, *Proc. Royal Soc. Med., Sec. Occup. Med.* Vol. 58, 1965:295-300.

1. Strength 2. Consistency 3. Specificity (mechanism: the same kind of tumor should be expected to appear in similar studies) 4. Temporality 5. Biological gradient (exposure or dose response – "the more the worse") 6. Plausibility (known phenomena and laws in physics, chemistry and biology) 7. Coherence 8. Animal experiments 9. Analogy

These criteria were applied to the issue of whether electric power lines could cause cancer or not. The analysis was made by thirteen American professors, including six Nobel laureates, in physics, chemistry and medicine[2]. The conclusion was that none of Hill's criteria was fullfilled.

Two reviews conclude as follows:

"The hypothesis that the epidemiological associations observed between 50/60 Hz EMFs and disease reflect a causal relationship is not supported by what is known about mechanisms."[3]

"Overall, the existing evidence for a causal relationship between RF radiation from cell phones and cancer is found to be weak to nonexistant."[4]

A recent publication regarding brain tumors and the use of NMT phones[5] is inconsistent with other publications including a retrospective cohort study of the incidence of cancer in all 420,095 users of cellular telephones during the period of 1982 to 1995[6].

Some previous key findings that have not been possible to replicate

A number of suggested molecular mechanisms for the induction or promotion of cancer as a result of EMF exposure have been tested. These mechanisms include the induction of oncogenes, signal transduction according to a calcium membrane leakage – protein signal pathway and DNA damage. Several studies along these lines originally reported positive associations with EMF exposure.

Looking into the rear view mirror, we now see that none of these initial findings can be replicated by independent investigators. The early

[2] Brief of amici curiae, Adair, Bloembergen, Bodansky, Cormack, Gilbert, Glashow …, in M. Covalt *et al*. Vs San Diego Gas and Light before Supreme Court of California, 1995.
[3] P.A. Valberg *et al*., Can Low-Level 50/60 Hz Electric and Magnetic Fields Cause Biological Effects? *Radiat. Res*., Vol. 148, 1997:2-21.
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reported EMF induction of the c-*myc* oncogene could not be replicated[7,8]. For example, earlier calcium membrane and signal transduction findings cannot be repeated at levels as high as 150 microtesla (50 Hz)[9,10]. The calcium leakage hypotehesis originally created headlines in newspapers such as Swedish Svenska Dagbladet (Dec. 4, 1995) and has also been covered by Swedish TV media. For the calcium membrane issue, it may in addition be mentioned that there has been a legal settlement regarding scientific misconduct (the so called “Liburdy affair” – a couple of key publications have partly been retracted but few formal details are available due to an agreement between those most involved and concerned).

For genotoxic findings (carcinogeneisis, mutations etc) it may be mentioned that some researchers are studying indirect endpoints such as DNA mass (as a surrogate for DNA breaks) which may lead to serious misinterpretations with methods such as the comet assay. For the genotoxic studies in general the criticism has been related to weakness concerning three basic conditions: independent reproducibility, consistency with the scientific knowledge base, and completeness according to basic data quality criteria[11].

An important replication study (which failed) regarding exposure of mice to 898.4 MHz and tumors was published as this text was being prepared[12]. Previous criticism included the exposure conditions, lack of exposure-response, statistics, the use of cancer prone animals, and that the animals that were still alive at the planned end of the study were assumed to be lymphoma-free, but were not proven to be so.

What is meant by possible risks?

An argument often used in the EMF controversy is that EMF is a possible cancer risk. While this is true, it also represents an unfortunate wording. Nothing can be excluded as a possible cancer risk – not even raspberry juice.

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[7] J.D. Saffer and S.J. Thurston, Cancer risk and electromagnetic fields, *Nature*, Vol. 375, 1995:22.

[8] G. Taubes, Another Blow Weakens EMF-Cancer Link, *Science*, Vol. 269, 1995:1816-1817.

[9] S.C. Miller and M.J. Furniss, Bruton’s tyrosine kinase activity and inositol 1,4,5-trisphosphate production are not altered in DT40 lymphoma B cells exposed to power line frequency magnetic fields. *J. Biol. Chem.*, 273, 1998:32618-26.

[10] H.E. Wey *et al.*, 50-Hertz magnetic field and calcium transients in Jurkat cells: results of a research and public information dissemination (RAPID) program study. *Environ Health Perspect.*, Vol. 108, 2000:135-140.

[11] McCann *et al.,*Genotoxic potential of electric and magnetic fields: an update. *Mutat. Res.*, Vol. 411, 1998:45-86.

[12] Utteridge *et al.*, Long-term exposure … mice to 898.4 MHz Microwaves … *Radiat. Res.*, Vol. 158, 2002:357-364.

Nevertheless, some careless scientists repeatedly exploit this fact and thereby indirectly create an unfounded public hysteria. It is therefore important to clarify this aspect to a larger fraction of the public (including some scientists!).

Conclusion

1. Inconsistencies in scientific findings. For epidemiological data: Many studies show no association. Weak associations may be due to statistical variations or other factors. 2. No exposure ("dose") responses 3. Experiments cannot be repeated by independent groups (many hypothesis have been tested) 4. No convincing "positive" genotoxic data 5. For low frequency fields ("50/60 Hz") and probably also for high frequency fields no physical mechanisms seem plausible.

The overall impression is that EMF does not constitute a genotoxic risk. If such a risk exists it is probably so low that it cannot even be detected and the distinction between these two alternative interpretations will therefore never be made.

Supervision of sunbeds in Finland

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Introduction

Sunbeds emitting ultraviolet radiation (UV radiation) are used for cosmetic tanning. UV radiation incontrovertibly causes skin diseases such as skin cancer and eye diseases. UV exposure from natural sun should be moderate and from sunbeds it should be avoided.

The aim of the supervision of sunbeds and tanning facilities is to ensure that they comply with valid safety requirements. The basis for the requirements is that acute effects such as sunburns will not occur and the yearly UV dose would not increase excessively.

Former regulations

Since 1989 the Finnish Radiation and Nuclear Safety Authority (STUK) has published guidelines for radiation safety of sunbeds. They have been mostly based on the European standard EN 60335-2-27 "Safety of household and similar electrical appliances - Part 2: Particular requirements for appliances for skin exposure to ultraviolet and infrared radiation". Only UV type 3 sunbeds have been accepted for cosmetic purposes in Finland. STUK made pre-marketing type inspections of sunbeds until the end of 1993. After Finland joined the EEA and EU, the supervision of sunbeds by STUK has consisted of market control and inspections at tanning facilities.

Present regulations

Since 1.5.2002 the Decree of the Ministry of Social Affairs and Health on the Limitation of Public Exposure to Non-ionising Radiation (294/2002) has been in force. This new Decree sets more detailed, binding public exposure limits of non-therapeutic/cosmetic ultraviolet radiation use. According to the Radiation Act (592/1991) the Decree is enforced and supervised by STUK. The requirements of the Decree are adopted in STUK's new guidelines "Radiation safety requirements and supervision of sunbeds".

In the Decree the main exposure limits of ultraviolet radiation for sunbed use are the following:

- the effective radiant exposure of ultraviolet radiation to the skin should not exceed 5 kJ/m^2 in one year
- the effective irradiance of ultraviolet radiation to the skin should not exceed 0,15 W/m^2 determined separately in the wavelength ranges less than 320 nm, and 320 - 400 nm, respectively, and the combined effective irradiance of both wavelength ranges should not exceed 0,3 W/m^2
- a person under the age of 18 should not be exposed to ultraviolet radiation of a sunbed except during an operation controlled by the medical doctor.

Earlier the maximum yearly dose was 15 kJ/m^2 (erythemally weighted) as it is in the European standard. However, the dose 15 kJ/m^2 would allow regular sunbed use. The average dose of one tanning session can be calculated to be about 230 J/m^2 (1). The dose 5 kJ/m^2 thus means about 20 tanning sessions per year with a typical UV type 3 sunbed. To reduce the risks of sunbed use, STUK's recommendation for yearly use of sunbed (maximum 10 sessions) is half of what the new formal decree would allow. The recommendation follows recommendations given by the WHO and the Association of Dermatologists in Finland.

According to the Decree also in the future only UV type 3 sunbeds are accepted for cosmetic use in Finland. The use of sunbeds shall be arranged so that the requirements presented in the standard EN 60335-2-27 will be fulfilled. If other than UV type 3 sunbed is to be used for cosmetic purpose, the operation must be done under the supervision of a professional with expertise in UV phototherapy.

The age limit is normative and it is based on the fact that children and young people have more sensitive skin than adults do. It should be said near the sunbed that it´s use is not re-commended for people under age 18.

Supervision

The supervision of sunbeds by STUK consists mainly of market control and spot checks of tanning facilities. In 1998, STUK launched a nation-wide survey of the use of sunbeds in Finland. Inspections were carried out in co-operation with municipal health officials. According to this survey it is estimated that there are about 700 tanning facilities and 1000 commercial sunbeds in Finland.

Deficiencies affecting the safety of sunbed users were discovered in nearly every tanning facility. Half of the sunbeds did not have an adequate timer for adjusting the length of exposure. Half of the sunbeds did not have warning signs indicating the health hazards of UV radiation and the necessary precautions required, and one quarter had no instructions at all. Only one third of the sunbeds had exposure schedules taking into account individual skin sensitivity, which is essential for correct UV dose. In most cases the UV levels were found to be within accepted limits for UV type 3 sunbeds. However, it was estimated that one tenth of the sunbeds have UV radiation levels which may exceed these limits.

The current practice (market control, spot checks and information of the public) has proven to be effective in the supervision of commercial sunbeds. In recent years the supervision of tanning facilities has also become essentially more effective, because municipal health officials have actively taken part in the supervision together with STUK.

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Communication and Public Information in Development and Maintenance of Radiation Safety

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Introduction

The natural source of Non-Ionizing radiation (NIR) in our environment is the sun. Artificial NIR is widely used in medical and industrial applications and also in households. The most important medical NIR applications are phototherapy, surgical lasers and magnetic resonance imaging. In industry NIR is applied in radio frequency heaters and laser printing. NIR has applications even in entertainment and cosmetics: High power lasers are used in light shows and sun beds emit artificial ultraviolet radiation (UVR) for tanning.

From the safety point of view the field is indeed very wide from harmless and common items to devices which can cause acute health hazards. The use of non-ionizing radiation needs neither any safety licence nor other kind of authorisation. The users of NIR devices varies from professionals in health care or lighting technology to those who only use them as a sideline in their main work like beauticians. Also laypersons use various NIR devices like microwave ovens, laser pointers and mobile phones.

Many NIR devices are used in applications where the user does not have the knowledge to understand the health risks associated with the device. Thus the only way is to educate the users by communication and public information.

Examples

Sometimes new kinds of NIR sources are adopted quickly in wide use. Reckless behaving or insufficient user instructions can cause hazardous exposure and even actual health hazards. Microwave dryers intended to dry wet concrete are examples of potentially hazardous novel applications. Couple of years ago problems arouse when damp damages of inhabited buildings were renovated by using high power microwave dryers. Microwaves were transmitted through the concrete walls to next apartments and staircases. Because those dryers are not objects of supervision, the safe use of them was achieved by an intensive information campaign. The campaign was targeted to building renovating companies and house managing companies. In the campaign safety recommendations (1) were delivered for the user of the dryers and after short period the use of dryers had settled to safe manner.

In the use of UVR for tanning the need of public information is most evident. This concerns both sunbathing and the use of sun beds. We have targeted the information to two distinct groups, to the general public and to enterprisers offering sun beds for customers in gyms and in beauty salons.

The aim is to make people aware of the risks of tanning and remain of sensible acting in sunshine. We also point out that tan is not a sign of health as still commonly believed. The use of sun beds is discouraged by correcting the two common misbelieves: 1) tanning by sun bed is safer than by solar UVR, 2) use of sun beds provides protection against solar UVR.

The public information is distributed by leaflets, media and Internet. Targeting a press release to women's magazines and other magazines of beauty and fitness the most enthusiastic group of sun bed users and sunbathers is attained. Nowadays we have achieved a reliable and high profile in media. Often newspapers and broadcasting companies make the initiative for interviews and offer visibility both in local newspaper and in national TV-broadcasts. Thus we can achieve a wide audience. It is essential to avoid too authoritative appearance. Instead we deliberately use a positive attitude and common speech and encourage people to use common sense. If possible, we put to limelight a young and preferably female researcher. Young females are heavy users of sun beds, and they may accept information more readily from a person with whom they can identify themselves. Women are also the target group when we promote people to protect their children against excessive sunbathing and sunburns.

The information targeted to gyms and in beauty salons reminds them of their responsibilities and provides advice and instructions how to fulfil the valid requirements. The information is distributed mainly by mail and via local health authorities. Media is also used to arouse the awareness of the entrepreneurs.

The Decree of Ministry of Social affairs and Health (2) gives upper limits for the UVR dose rate of sun beds. The requirements for the instructions and information available to users of sun beds are given in the Decree and described in a European standard (3). These requirements apply to those who offer sun beds for customers' use. STUK supervises the sun bed facilities in co-operation with municipal health authorities. Supervision and inspection can never reach all the facilities. The site inspections and their after-care have demonstrated that entrepreneurs have enormous difficulties to provide all the required instructions and safety information for the customers. The importers of the sun beds often offers incomplete information material but they provide often good looking glossy posters where extracts of adequate information is mixed with misleading advertising. If the entrepreneurs tries to add all the missing instructions the result is often a mess of photocopies and various printouts plastered on the walls. This is only confusing and frustrating to the sunbed users.

Confronted with the reality we decided to solve the problem by doing the work ourselves. An elegant poster was designed to present all the required guidance and safety information required by references 2 and 3. The lay out of the poster was designed to make the text easy to read and make the poster appearance so aesthetic and classy that the gyms and beauty salons would accept to hang it on their walls. The poster is delivered free to sun bed facilities by mail and during site inspections. The poster is also available for printing in our WWW-site.

Hopefully, providing this free poster to every sun bed facility we can improve the safety of the sun bed user far more effective than by supervision and inspections.

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New laser classes

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Abstract

By an up-dated international standard (IEC 60825-1 + Amendment 2) on laser safety some new laser classes are introduced. The new set of laser classes consists of 1, 1M, 2, 2M, 3R, 3B, and 4. This is a result of intense discussions in the committee and was laid down in 2000, slightly adjusted 2001. The previous classes 1, 2, 3A, 3B, and 4, established since more than 25 years, are partly abandoned. This paper compares the new classes to the old ones.

Introduction

The IEC committee TC76 has since many years dealt with laser safety, exposure limits and laser classification. The exposure limits (maximum permissible exposure, MPE) form an important part of the standard IEC 60825 which has been up-dated and amended several times since 1984. The exposure limits are expressed as functions of, among others, wavelengths, time patterns, and retinal image sizes. Thus their use is not too easy for persons who have not particularly considered the matter. No higher mathematics is involved, but there are a number of things to think about.

To make things easier for the users, a system of classification of lasers and products containing lasers was invented. The user then easily gets an idea about the potential risks and how the work on safety should be designed. Since some years, also light emitting diodes (LEDs) are included within the scope of the standard.

Discussions within the committee on new laser-classes have been going on for several years. The main reasons have been that one did not want to be unfair to the LEDs and also to approach the national system in USA, which since many years has been different from the rest of the 'IEC world'. After meetings in Mishima (Japan) 2000 and Kista (Sweden) 2001, the new laser classes have taken their new shapes. The philosophies behind the new classes can be described as follows, but we will first look upon the old classes:

Old classes

Class 1

Lasers (and LEDs) are harmless even to long term exposure. Either the lasers are so weak that they are not capable of producing any health effects whatsoever, or we deal with products that contain lasers, which might be of higher classes, but are safely built-in and interlocked in such a way that no harmful radiation can be emitted from a product at normal use. The maximum radiation power or the maximum pulse energy in the class is directly related to the exposure limits. The upper class-limits of these parameters are of course strictly defined, but the complexity of the exposure limits does not permit a simple description.

Class 2

This class only contains lasers that emit visible radiation, defined in the standard as radiation in the wavelength interval 400 - 700 nanometres. (It is possible to see radiation of wavelengths for instance 380 nm or 750 nm as well, but in these spectral regions the relative vision response of the human eye is too low to guarantee a sensation of glare.) The sensation of glare is important. If an unprotected eye should be hit by a class 2 laser beam, natural mechanisms are provoked to avoid further exposure i.e. the eyelid closes. These mechanisms are fast enough (< 0.25 seconds) to prevent overexposure of the retina. For CW-lasers the upper class limit is 1 mW. CW stands for "continuous wave", and such lasers emit radiation with constant power from switched on to switched off.

Class 3A

The class contains lasers emitting visible radiation (400 - 700 nm) as well as invisible radiation (< 400 nm, i.e. ultraviolet, or > 700 nm, i.e. infrared). In the visible case the laser may totally emit 5 times the class limit of class 2, that is 5 mW for CW-lasers, but the beam must be expanded so the power density is at most 25 W/m^2. This implies that the retina of the naked eye can not receive more than about 1 mW, which is equivalent to the exposure by a class 2-laser. However, if the beam is viewed through binoculars or other collecting optics more than 1 mW may enter the eye.

For invisible radiation the philosophy is analogous but here 5 times the limit of class 1 is allowed and, simultaneously, the condition for the beam to be expanded in such a way that the MPEs can not be exceeded, provided that no collecting optics are used.

Class 3B

As soon as we exceed the class limits for class 1 (invisible) or class 2 (visible), we run right into class 3B, if we are not in the lower end and the beam is expanded according to the conditions for class 3A. Class 3B-lasers are considered to be hazardous for direct exposure to the eye and, at least in the upper region, to the skin. However, it is not hazardous to view diffuse reflexes, i.e. reflexes from a matte surface. The upper limit for CW-lasers is 0.5 W.

Class 4

This class contains lasers that can be extremely dangerous. Direct exposure is dangerous for eye and skin and even diffuse reflexes may be hazardous at least under some circumstances. The class has no upper limit.

New classes

Class 1

is in principle unchanged.

Class 1M (new class)

This class contains lasers, emitting visible or invisible radiation (or both), with a total power or pulse energy that exceeds the class 1 limit, but the beam must be expanded so the MPEs are not exceeded. The philosophy is similar to the former class 3A but the cap (five times the class 1 limit) is removed. Thus lasers in the class are harmless provided no collecting optics are involved. M stands for magnifier.

Class 2

is not changed.

Class 2M (new class)

The class contains lasers emitting visible radiation (400 - 700 nm). The limit of class 2 is exceeded, but the beam must be expanded to allow an exposure of duration 0.25 seconds. This means for CW-lasers a maximum power density of 25 W/m^2, as we recognise from the old class 3A in the visible region. Also here the cap (five times the class 2-limit, i.e. 5 mW for CW-lasers) is removed.

Class 3R (new class)

This class contains lasers in the lower end of the old class 3B. The class limits are defined as five times the upper limits of class 1 (invisible) or five times the limits of class 2 (visible), regardless of the geometric shape of the beam, unless the beam is expanded to meet the requirements of class 1M or 2M. It is possible to exceed the MPE-values, but they have inherent safety factors, so lasting damages on eyes or skin are extremely unlikely in reality, should a short term human exposure occur. R stands for restricted.

Class 3B (modified class)

This class is similar to the old class 3B but the lasers in class 3R are removed.

Class 4

is not changed.

Consequences

A crucial problem with the old system was that many LEDs were overestimated when it comes to evaluation of hazards and classification. It was difficult to create measurements procedures for classification taken into account the fact that most LEDs emit radiation from an expanded surface forming an exposed area on the retina rather than a point. The radiation fields are also strongly divergent. Earlier many LEDs were classified as class 3B. The new classes 1M and 2M give possibilities to more adequate classifications in better agreement to reality. Of course this was applauded by the manufacturers, but it is also advantageous from a workers protection point of view since otherwise the respect for "real" 3B-lasers would have declined.

The old class 3A is completely discarded. All such lasers are either put into class 1M or 2M. In addition, other lasers than former class 3A fit into one of these classes since the cap is removed.

The national system in USA differs from the IEC system. In USA there is a class 3A consisting of lasers in the visible region with a maximum power of 5 mW (CW). The former additional 3A condition, expanded beam, was abandoned a number of years ago. This has caused some confusion, since many products, for instance the majority of laser pointers, are labelled according to the USA system, i.e. 3A. The correct classification in IEC parlance is 3R.

Posters

6a Dosimetry in Phototherapy

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Introduction

Artificial ultraviolet radiation (UVR) is used in phototherapy, the medical treatment of various skin diseases like psoriasis and atopic eczema. Both solar and artificial UVR exposure have also carcinogenic effects. Long term phototherapy with high cumulative UVR doses is shown to be risk for squamous cell carcinoma (1,2).

Various biological effects are strongly dependent of the wavelength of the UVR. UV-B-radiation with short wavelength (280 - 320 nm) is more effective to cause sunburn of the skin than UV-A -radiation with longer wavelength (320 - 400 nm). The correlation between wavelength of UVR and the tissue sensitivity is described by action spectra. An action spectrum is a mathematical model describing the efficiency of UVR of different wavelength for producing a certain biological phenomena (3). Specific action spectra have been determined for the most important health effects like erythema and skin cancers as regard squamous cell carcinomas and basal cell carcinomas. (4,5).

Phototherapy devices

In most of phototherapy devices the UVR sources are fluorescent tubes. There is a wide selection of lamps producing different UVR spectra.

Phototherapy devices are classified to three main categories by the UVR spectrum (Figure 1):

- PUVA-devices are equipped with UV-A-lamps and they are used with certain photosensizing drugs, psoralens, for PUVA-treatments (PUVA = psoralen + UVA);
- UV-B-devices are equipped with broadband or narrow band UVB-lamps;
- SUP-devices. SUP is an abbreviation of Selective UV-phototherapy;

SUP-spectrum lies between spectra of UV-A and UV-B-lamps and it is commonly used for phototherapy in Nordic countries. In Germany SUP refers to treatments with UV-B-lamps.

Devices intended for whole body treatments are mostly cabins or cubicles where lamps are placed in vertical panels surrounding the patient. Sunbed shaped devices are also used. In local treatment of hands and feet a typically phototherapy device is a small size panel installed in an adjustable stand.

Dosimetry

The concept of the "dose" in photobiology and in photomedicine differs significantly from the one in radiobiology. The UVR dose is defined as the radiant exposure on the surface (3). This concept does not take account which proportion of the UVR is absorbed in the tissue and which is reflected or transmitted.

Traditionally phototherapy has based on assessment of the skin erythema between exposures. The redness of the skin has been used as an indication to shorten or lengthen the irradiation times. There has been no other means to assess the treatment doses.

The dose rate of phototherapy devices can be assessed by a simply hand held UVR-meter. When the dose rate is known it is possible to determine the UVR dose delivered in each treatment exposure as well as the cumulative dose of a patient.

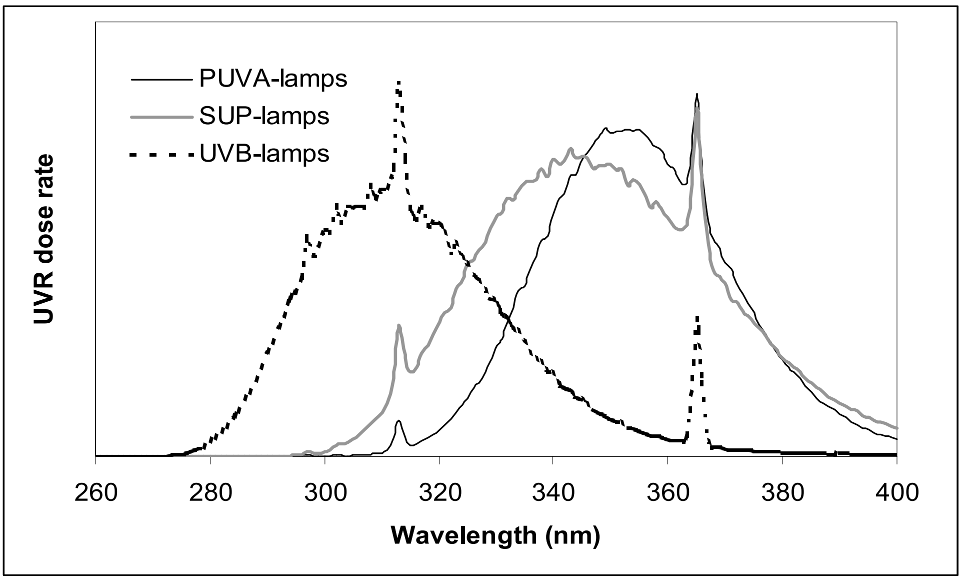

Figure 1. UVR spectra of three different kind of UV-lamps intended for medical purposes. The spectra are scaled to their maximum values.

In these cases treatment regimens can be planned and carried out by the means of UVR doses instead of exposure times. Nowadays some of the most advanced phototherapy devices have an intrinsic meter to monitor the intensity of UVR and an automatic system to adjust the length of exposure time when the dose rate changes.

It is essential that the UVR-meter is calibrated against the spectrum of the measured UV-lamps (6,7). An uncalibrated or inadequately calibrated meter may cause a gross error in the measurement result.

When the treatment regimens are based on the actual UVR doses it s possible compare the results of exposure regimens in different clinics (8).

The UVR dose rate of phototherapy device decrease when the lamps age. To maintain the intended treatment regimens the exposure times should be lengthen respectively to compensate the lower dose rate. On the contrary, when aged lamps are replaced with a fresh lamp set the dose rate is significantly higher. This may cause overexposure and skin burns if exposure times are not shortened appropriately.

Clinical practice in Finland

Although more than half of the phototherapy clinics in Finland have an UVR-meter they are seldom used for dosimetric purposes. In some cases regular dose rate measurements were carried out by service men only to monitor when it is time to replace a lamp set but the measurement results was not utilised in the treatments. There is also a wide variation between the clinics in making the decision when it is time to replace the aged lamps by new ones. Only in few clinics treatment regimens are based on UVR doses, the majority still follows time-based regimens. Thus, the recording of each patient's cumulative UVR doses has not came in practice (9).

Recommendations

Dermatologist in Britain and In U.S.A have given recommendations to keep records of the UVR dose of the patients (10,11). Similar recommendations is also given in Finland by STUK and the National Agency of Medicine (7,12). Measurements are mandatory in Sweden (SSI FS 1998:2). The main points of the Finnish recommendations are following (7):

- The personnel should have sufficient training to operate phototherapy devices.
- Phototherapy devices should have written instructions for use.

- The dose rate should be monitored by a calibrated UVR meter.
- Records should be kept on matters affecting to the UVR doses of phototherapy devices, i.e., the type of the lamps and dose rate.
- UVR doses of each patient should be recorded and treatment doses should be kept as low as convenient for the treatment.
- Patients' eyes shall be protected always and also critical skin areas should be covered if convenient.
- A regular review of patients with high UVR doses is recommended to detect malignant skin lesions.

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6b Dosimetry associated with mobile phone fields

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Concern about electromagnetic fields produced by mobile phones and the effects of the fields on human health have recently been under continuos debate among citizens, media and scientists. This concern is justified, even though the effects on health under the thermal threshold have not been established, because the use of mobile phones is so widespread nowadays.

To evaluate the health risk of mobile phones, reliable and repeatable biological studies must be conducted. Studies are conducted with various methods (epidemiological, human, animal, cell culture). Firm conclusions about the effects of radiation can only be made, if the results from several well-conducted studies coincide. Results from earlier studies have some contradictions. This is partly due to the fact that the equipment and methods, used in exposure determination, have been insufficient. To ensure high quality studies, the dosimetry must be as good as possible. The SAR (Specific Absorption Rate) must be precisely known and its' distribution as uniform as possible. This has been the key idea of the work done in the Non-Ionising Radiation Laboratory of STUK.

The SAR evaluation is based on Maxwell's equations, which are calculated numerically, because in the case of a real system the geometry is so complicated that analytical equations cannot be solved. The numerical calculations can be done with some computer-based algorithm (e.g. FDTD, FEM). In finite difference time domain (FDTD) algorithm, first some impulse is fed to the three dimensional computational space and then all the field values (in every voxel) in space are calculated from time step to time step until a steady state is achieved. The SAR distribution can be calculated from the E-field distribution from following equation:

$$SAR = \frac{\sigma E^2}{\rho} = \frac{c\, dT}{dt}$$

where σ is conductivity of material, E is the RMS (Root Mean Square) value of Electric field, ρ is density of material, c is specific heat, and dT/dt is the temperature rise. Computer based SAR evaluation should always be verified with measurements. This can be done for example with E-field measurements in air or tissue with miniature E-field probes (Narda or SPEAG) or with temperature rise measurements with small temperature sensors (Luxtron or Vitek).

The various aspects of dosimetry for mobile phones have been studied in our laboratory. We have developed a set of calibration waveguides for the calibration of E-field and temperature probes used in the measurement of E-field in air and liquids. This ensures that the fields and thus the SAR can be measured correctly, which is basis for reliable SAR-determination. The calibration uncertainties of waveguides (450, 900, 1800, 2000, 2450 MHz) are one of the lowest in the world and the results have been verified with intercomparison with other high quality laboratories. Other important factors in SAR measurements are the correct permittivity values of liquids used (e.g. cell culture liquid, different phantom liquids). Permittivity values can be measured with TEM-line sensor method developed in collaboration with Radiolaboratory of TKK (Helsinki University of Technology) and STUK. That is the most accurate method to measure the dielectric constant and conductivity of liquids around mobile phone frequencies (900 and 1800 MHz). With the correct permittivity values and calibrated probes it is possible to measure the SAR in phantoms, simulating for example human head or animal, or in cell cultures.

The SAR distribution of a mobile phone can be studied either numerically with computer simulations and real shaped models of a human head or experimentally with E-field measurements in liquid phantoms, simulating the electrical properties of average brain. An E-field scanning system (CADSCAN) has been developed by STUK for measurements of E-fields in various phantoms. Even though the CADSCAN can be used for mobile phone tests, it does not satisfy the positioning requirement (< 1mm) of the European Standard (CENELEC EN 50361). Therefore we are now installing a commercial robot based E-field scanning system (SPEAG), where the moving accuracy is 0.1 mm. The system should be ready in operation at the beginning of year 2003. With the system SAR distribution, emitted by a GSM phone, can be measured accurately and the comparisons to international (ICNIRP) or national (STM) guidelines can be made. The new system will be used for research and the market control of mobile phones.

In STUK two versions of a cell culture exposure chambers have been developed for mobile phone studies at 900 MHz. Both systems are water-cooled waveguide exposure chambers, allowing higher SAR exposure, when compared to normal air-cooled systems, due to the efficient heat removal. The higher SAR exposure, without the thermal effects, is essential to elucidate non-thermal effect dose relationship. The older system was constructed five years ago and

Figure 1. Parallel plate chambers for exposing rats to 900 MHz radiofrequency fields. The plastic rat cages are inside the chambers under the drinking vessels. The fresh air for rats is circulated from the tubes in the centre of each chamber. The electromagnetic field is produced with a real GSM-phone and amplified with power amplifiers and controlled with PC; all these in centre of figure.

the newer one two years ago. Both systems are of resonator type; at the older version the cells are located near the magnetic field maximum and E-field is parallel to the plane of medium. At the newer version the cells are placed at the E-field maximum, which is perpendicular to the plane of medium resulting in a more uniform distribution. The variation of SAR in whole dish volume is under ±30%. This can be compared to older version where the SAR decreases at the edges of cell dish to about 10% of the maximum value at centre of the dish. The dosimetry and exposure determination in cell culture studies have been done with FDTD simulations and they are verified with E-field and temperature rise measurements.

We have also developed exposure systems for animal studies (mouse, rat), which have been conducted in Kuopio University. A waveguide exposure chamber for 24 mice was developed and in that system the animals are restrained to plastic tubes during the exposure. For rats, nine exposure chambers each of which has 24 rats were developed and the dosimetry was conducted in collaboration with VTT (Technical Research Center of Finland). The rat chamber is a parallel plate waveguide system shown in Figure 1, where a radial TEM wave propagates from a centre feed outwards. The rats are located at cages, which are located near the outer circle of the chamber. Rats can freely move in their cages, which makes the SAR determination quite difficult because the rat absorbs the energy differently depending on its orientation and place. The benefits of rat system are free movement of rats, no stress, less work for personnel. These benefits are gained at the cost of increased uncertainty of dosimetry.

SESSION 7. RADIATION IN MEDICAL USE

Medical use of ionising radiation – challenges for the third millennium

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Introduction

From the very beginning after its discovery ionising radiation has been in beneficial use for health care. But even the drawbacks showed up very early: only a few months after Röntgens discovery reports were published on patients who got severe skin damage after fluoroscopy with x-rays. This finding of the adverse effects was soon turned into something positive: ionising radiation could be used for treatment of cancer. In 1928 radiologists took the initiative to the foundation of what later became the International Commission on Radiological Protection, ICRP.

Medical use of ionising radiation is giving by far the largest contribution to the radiation burden of the global population from artificial sources, on average 0,3 mSv per year and inhabitant, excluding doses from radiation treatment. In the Nordic countries this dose is approximately 0,7 mSv. This isn’t a problem by itself. The total benefit is exceeding the total radiation risk with large margins. But the margins could even be larger. Methods for examinations and treatments have often a potential for improvements, meaning that the medical effect can be obtained with a lower dose to the patient. In certain circumstances the examination does not contribute to the further treatment of the patient or to her/his well-being and is then regarded as not justified. The huge challenge we are facing depends among other things on the extreme fast technical development which enables exposures of a magnitude that we haven’t seen before and applications we only could dream about. There is a risk that the motto *do what is possible to do* is followed instead of *do what the individual patient needs.* This presentation addresses the possibilities, but also the dangers that medical use of ionising radiation in medical care is facing in the new millennium, or at least in its first years.

Trends and development

X-ray diagnostics

The number of x-ray examinations has been rather constant during the last 20-30 years. But an enormous development went on that has influenced the radiation doses in both directions. With the introduction of the image intensifier in the 1960s and rare earth screens in the 1970s the radiation doses for conventional x-ray examinations could be reduced considerably. In addition the number of high-dose examinations such as barium meal and barium enema decreased steadily being replaced by gastroscopy and endoscopy. Ultra sound and magnetic resonance examinations have also substituted a number of x-ray examinations.

On the other hand computed tomography has experienced a tremendous development since its introduction into clinical use in 1972. Computed tomography accounts today for less than ten percent of all x-ray examinations but for 30-50 percent of the collective dose from x-ray examinations. And the fields of application are increasing more and more. Within a few seconds a detailed 3-dimensional image of a multi-trauma patient can be produced, and with virtual coloscopy we can make a journey through the intestine where lesions and other changes can be studied in detail.

The use of digital x-ray technique has also increased outside the x-ray departments: cardiologists, neurologists and urologists are using x-rays for guidance in therapeutical measures, for the so-called interventional procedures. This trend is most likely to go on in the foreseeable future.

Nuclear Medicine

Whereas x-ray diagnostics is mostly used to reproduce images of different organs or tissues in the body, to describe anatomical conditions, the strength of nuclear medicine is to reflect dynamic processes (how the organs are working). Substances with radioactive isotopes are administered to the patient, intravenous, by indigestion or inhaling. The travel of these radionuclides through the body is monitored by measuring the radiation emitted by means of suitable detectors. Uptake of iodine in the thyroid, mapping of the brain activity and determination of the lung function are some examples of nuclear medicine procedures. Tumour searching molecules are used for both diagnostic and therapy. Although new radioisotopes and new methods have been introduced the total number of nuclear medicine examinations and treatments has been rather constant during the last 10 to 15 years.

Radiation therapy

Soon after the discovery of x-rays the radiation treatment of tumours started. In the beginning this was only possible for superficial tumours because of the low radiation energy. In lack of reliable physical measuring methods the quantity *skin erythematic dose* (HED) was introduced. HED was determined with the physician himself as a detector. He (or she) irradiated his (or her) own arm and determined the irradiation time necessary to induce redness.

Radium was the most frequently used radioisotope for therapeutical purposes before the turn of the last semi-century. With the introduction of nuclear reactors Co-60 became available, and cobalt equipments were replacing the x-ray therapy equipments. Soon followed the first accelerators and today they have replaced almost all cobalt equipments.

Challenges for radiation protection

X-ray diagnostics

Several aspects in the recent development of x-ray diagnostics are likely to have great importance for radiation protection issues. There is a risk that the development is going too fast and that radiation protection is falling short. It happens easily that the new achievements are so overwhelming that nobody is caring about eventual side effects or is questioning the need of this method. Adequate education and training of the staff, in the first hand of the radiologist, is of utter importance but unfortunately frequently neglected.

Radiation protection in x-ray diagnostics can be summarized with the statement that all examinations shall be justified and optimised. Some aspects in the latest development can be pointed out where special attention is needed. X-ray equipments are becoming more and more powerful, and the natural limitation of the exposure that is associated with the use of film (over-exposed films cannot be used for the diagnosis) is absent with digital systems. An increase of the radiation dose in excess of the level necessary for a safe diagnosis is hardly visible in the images – what happens is maybe a slight reduction of the noise, and who is complaining about that? With modern computed tomography equipment the whole body can be scanned within a few seconds giving a three-dimensional image with high resolution – is that always needed for the patient's well being?

Interventional procedures, where therapeutic procedures are performed assisted by fluoroscopy have been developed more and more. Examples are the reparation of injured or obstructed vessels, the ablation of nerves in the heart for the elimination of the disturbance of the heart rhythm, implantation of pacemakers. Frequently these procedures are combined with long fluoroscopy times, especially when complications occur with e.g. the introduction of the catheter. Acute radiation injuries such as slowly healing ulcerations in the skin have been reported for the first time again since the time when x-ray diagnostics still was in its infancy. This could have been avoided if the equipment had been used in a correct manner and if the staff had have better education and training in radiation protection, radiation physics and x-ray technique.

Nuclear medicine

Within nuclear medicine two areas can be mentioned that deserve special attention. One is the *sentinel node procedure,* where the purpose is, e.g. for a breast cancer patient, to identify those lymphatic nodes that are in direct connection with the tumour, so that the removal in the lymphatic system can be restricted to what is necessary. A radioactive substance is injected directly into the tumour and by measurement with an external detector those lymphatic nodes with an uptake are identified. There might be a radiation protection problem because the practice is carried out in a new environment outside the nuclear medicine department (operation theatre) with personnel not educated in radiation protection (surgeons). Education and training and establishment of radiation protection rules are crucial for the practice to be run satisfactory.

The other area is PET (positron emission tomography). Also here a radioactive substance is administered to the patient, a β^+-emitter. The two 511 keV annihilation photons are measured in coincidence and so the distribution of the radioactive substance is localised in the body. A three-dimensional image is created which is used among other things to study the glucose metabolism, the cerebral blood flow and the blood flow in the coronary arteries. For tumour diagnostic F-18 FDG is increasing with the increased availability of this method.

Radiation protection problems may be present for the staff during the process of manufacturing and preparation of the radiopharmaceutical and in connection with the administration, and also for the patient herself. Very high activities are dealt with and the energy of the radiation is high. One special problem is the finger dose from the preparation and administration. The radioisotopes used have short half-lives. Standard gamma cameras can be modified such that they can be used for PET examinations, and that will lead to a spread of this technique to more hospitals. Longer distances between the accelerators where the isotopes are produced and the hospitals are leading to longer transport distances and a need for higher activities.

Radiation therapy

Radiation therapy is a balancing act: The tumour must receive a sufficient high radiation dose in order to kill all cancer cells whilst the surrounding healthy tissue must be spared as much as possible. The development of diagnostic methods (computed tomography, magnetic resonance, PET), of computer power and of accelerators with the possibility of tailor-made adjustments of the beam has given better and better tools to the therapist for this purpose. With the latest technique IMRT (intensity modulation radiation therapy) it is possible, at least in principle, to choose any incident radiation field with respect to direction, field size and radiation intensity. A difficulty is to determine the real dose distribution that is achieved of the thin scanning beam and to determine the anatomical landmarks with respect to location and attenuation properties.

Some people with certain hereditary changes in the genes have higher radiation sensitivity than the average. Research is going on with the purpose of obtaining more knowledge about this phenomenon and to use this knowledge for individual adaptation of the therapeutical dose: lower to those with high sensitivity, higher for the others.

Intravascular brachytherapy is a practice where competence is needed within x-ray diagnostics, radiation therapy, and to a certain extent also in nuclear medicine, and both with respect to clinical and to physical-technical aspects. Of importance is the establishment of a well-designed organisation that is ensuring that the competences needed are present and that the co-operation is performing well.

Conclusions

Ionising radiation has been at the healthcare's service for more than a century and there are no indications that this will change in the foreseeable future. In total the benefit is outweighing by far the deleterious effects caused by radiation. But there is a large potential for improvements. In some applications the radiation dose can be reduced without jeopardising the medical objective, in other cases alternative methods without the use of ionising radiation could be a better option. In some cases the use of ionising radiation is totally unnecessary if e.g. the result of the diagnosis is of no significance at all for the patient. From the diversity of application areas some can be mentioned that are entailing special requirements on radiation protection in the near future.

Generally the development is going more and more towards complex equipment and methods. The radiation protection conditions may be difficult to grasp when dealing with advanced technology equipped with many automatic functions, and it may be difficult to choose the correct settings or to adapt parameters for the actual application. Efforts are necessary concerning the equipment with e.g. standards prescribing that the settings must be flexible for adaptation to the various clinical situations, and user-friendly documentation of the equipment must be provided. On the user side the major demands are education, education and education. This is of special relevance when advanced equipment is located outside the traditional clinics in an environment with lacking radiation protection competence among the users – a trend that can be observed more and more.

Justification is another area of interest. The new examination methods as e.g. the new applications of computed tomography must be scrutinised from the patient's perspective. Is really the examination from the top to the toe to the benefit for the patient, or is it performed on flimsy grounds? This question must be asked and answered for all examinations that are planed, but it is especially important for new procedures with new technology in order to avoid the introduction of a practice that is inferior from a radiation protection point of view.

Continuous optimisation can lead to large dose reductions. A tool for that purpose in x-ray diagnostics is the introduction of the concept diagnostic reference levels. At the least a dose reduction of five percent can be expected. In the Nordic countries this would imply a dose saving of 800 manSv, every year.

The discussions on actual and potential problems for the present and the future use of ionising radiation in health care could give the impression that these practices are full of problems and should be questioned from a radiation protection point of view. That is completely wrong. Even if much can be improved there is a huge benefit for millions and millions of patients. Many lives are saved by the diagnostic and therapeutical procedures. In a global perspective the major issue is not that too many persons are exposed to medical exposures but too few: the availability of medical practices involving ionising radiation is concentrated to the developed countries. An expansion of these techniques to the developing countries would imply a large step forward for the public health in those countries.

Nordic working group on x-ray diagnostics – Practical implementation of the directive on medical exposures in the Nordic EU countries

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Introduction

The EU directive[1] on medical exposure, 97/43/EURATOM (referred to in the following as *MED*) imposes new requirements on hospital departments using ionising radiation for either diagnostics or treatment of illnesses. The directive was approved on 30 June 1997, and the member states were obliged to implement the requirements into national legislation before 13 May 2000.

The implementation of a directive of this kind is a complicated process requiring time as well as other resources. The Nordic EU countries (Sweden, Finland and Denmark) must comply with the rules in *MED*, while this is not the case for Norway and Iceland as EFTA (European Free Trade Association) members, since the agreements between EFTA and EU does not cover the EURATOM treaty.

The issues that have to be addressed in the national legislation are justification, optimisation, responsibilities, procedures, training, equipment, special protection during pregnancy and breast-feeding, and potential exposure. A central aspect in *MED* is the requirement for quality assurance programmes to be established in radiological departments (and in other departments employing ionising radiation). A change of this magnitude in legislation requires adjustments in the routines of the individual departments. The staff in each department needs to prepare and follow procedures and instructions for daily work and also participate in day-to-day quality assurance.

A considerable burden has also been laid on the radiation protection authorities in the member states, first in the process of transposing *MED* into national law or regulations, and secondly in guiding the process of practical implementation. Here we will describe how the individual Nordic EU countries have chosen to implement *MED* in national legislation and how far the process of complying with the requirements has come so far. Although Norway and Iceland are not required to follow *MED*, it is still interesting for comparison to include the situation in these countries. We will not cover all aspects of *MED*, instead we have chosen specific points that we think are the most interesting. The scope is limited to diagnostic applications of x-rays, excluding dental, therapy and nuclear medicine applications.

Regulations implementing the directive

MED was implemented into the legislation of all of the Nordic EU countries[2,3,4] before May 2000 as required.

Furthermore, new radiation protection laws were issues in Norway[5] in May 2000 and in Iceland[6] in April 2002, both implementing parts of *MED*.

Responsibilities

According to *MED*, the member states shall ensure that any medical exposure is effected under the clinical responsibility of a practitioner *(Article 5 (2))*, where a practitioner is defined as *(Article 2)*: a medical doctor, dentist or other health professional, who is entitled to take clinical responsibility for an individual medical exposure in accordance with national requirements. However, the practical aspects for the procedure may be delegated to other professionals *(Article 5 (3))* (these professionals are referred to in the following as *entitled individuals*).

National requirements thus had to be established on who can serve as responsible for an x-ray facility, and these requirements differ among the Nordic countries. Sweden has the strictest requirements, requiring the person responsible to be a radiologist, while Finland and Denmark are less strict, allowing also physicists and chiropractors, respectively. A complete list of the formal educational requirements is given in Table 1.

Differences also exist in the requirements to the *entitled individuals*. Finland and Denmark have set up specific requirements to the education of

Table 1. Educational requirements to the practitioner responsible for the x-ray facility.

Educational requirements to the practitioner responsible for the x-ray facility:	
Sweden:	Radiologist
Finland:	Physician or physicist
Denmark:	Radiologist, physician with relevant supplementary education or chiropractor
Norway:	Physician or chiropractor
Iceland:	Physician or chiropractor

Table 2. Educational requirements to *entitled individuals* (see text).

Educational requirements to *entitled individuals* (see text):	
Sweden:	None (under responsibility of the responsible radiologist)
Finland:	Physician, chiropractor or radiographer
Denmark:	Physician, chiropractor, radiographer or nurse
Norway:	None (under responsibility of the responsible doctor/chiropractor)
Iceland:	Physician, chiropractor, radiographer, nurse

the *entitled individuals*, while Sweden allows anyone to photograph under the responsibility of the responsible radiologist. The formal educational requirements are listed in Table 2.

Medical physics expert

A new aspect in the directive was that all facilities for medical exposure of patients must have a medical physics expert involved. In diagnostic radiology departments, the medical physics expert should be involved as an advisor in optimisation including patient dosimetry, in quality assurance, and in matters of radiation protection *(Article 6 (3))*. This has not been a tradition in all countries, and where this is a new requirement, time is required for further education, establishment of interpersonal networks and collection of expertise.

Involvement of medical physics experts in therapy was common in all of the Nordic countries, but involvement of medical physics experts in diagnostic radiology had only been a tradition in Sweden and they also had a formalised education. Finland now also has a medical physics degree, while the education as medical physics expert in Denmark is based on an individual study program for each candidate. In Norway and Iceland the medical physics expert is not defined as a health profession in the legislation, and no formalised educational system exists for this group.

Table 3. Estimated number of medical physics experts in diagnostic radiology per 1 mill. inhabitants.

Estimated number of medical physics experts in diagnostic radiology per 1 mill. inhabitants				
Sweden	Finland	Denmark	Norway	Iceland
10	2	4	0	0

Equipment

MED requires an up-to-date inventory of radiological equipment at each department to be available to the radiation protection authorities *(Article 8 (2))*. However, since national inventories have existed for at least 10 years in all of the Nordic countries except Norway, this requirement was already implemented.

To ensure that the equipment performs adequately, acceptance testing must be carried out before first use for clinical purposes *(Article 8 (2))*. In a transitional period, acceptance tests must also be carried out on all existing equipment. This is a process requiring some time, since it is simply not possible to test all equipment at the same time. The Nordic ambitions for acceptance tests were outlined in a Nordic report in 1999[7], and the practical implementation is referring to this report. By now, the process is finished in Sweden, nearly finished in Iceland, but still some remains to be done in Finland and Denmark, as seen in Table 4. In Norway the legal basis for imposing such tests is still not clear, that means it is up to the hospital to prioritise the purchase of necessary measuring devises and training of local staff for such tests.

As part of their quality assurance programmes, the Nordic countries have introduced constancy tests, i.e. testing on a frequent basis that all the key parameters in the picture producing process remain constant. Implementation of constancy testing has also required some time, since the regular staff in the

Table 4. Estimated status of implementation of acceptance and constancy testing.

	Acceptance test	**Constancy tests**
Sweden	100 %	100 %
Finland	20 %	100 %
Denmark	70 %	50 %
Norway	10%	20%
Iceland	95%	100 %

department mainly carries this out, and the necessary education had to be completed first. As seen in Table 4, constancy test are by now fully implemented in Sweden, Finland and Iceland.

Inspection

To ensure that the departments comply with the regulations, *MED* requires the national authorities to set up a system of inspection *(Article 13)*. The systems set up divides into two groups: Sweden and Denmark have chosen an approach with audit of the quality assurance system and then spot testing of selected issues, while Finland inspects the practices in the department. The system in Iceland is close to those in Sweden and Denmark, but with a more thorough inspection of equipment. In Norway the focus is on control of equipment. The interval between inspections lies in the range of 2 – 5 years, except in Norway where the range is 5 – 10 years. The systems of inspection are summarised in the table below.

Table 5. Inspection systems set up by national radiation protection authorities.

	Type of inspection	Interval between inspections
Sweden	Audit, spot tests	3-5 yrs
Finland	Inspection of practices	2-5 yrs
Denmark	Audit, spot tests	2-3 yrs
Norway	Control of equipment, spot tests	5-10 yrs
Iceland	Audit, control of equipment	2-3 yrs

Conclusions

The Nordic EU countries, Sweden, Finland and Denmark, have progressed a long way towards full practical implementation of the Directive on medical exposures. Some notable and interesting differences in the national requirements exist, but the trends are very similar. The pace of the practical implementation also differs somewhat with Sweden seeming to be in the front at the moment. In the aspects covered here, except for medical physics experts Iceland seems to follow the directive quite closely.

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Nordic working group for medical x-ray diagnostics: Diagnostic reference levels within x-ray diagnostics – experiences in the Nordic countries

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Introduction

Medical x-ray diagnostics is one of the few applications of ionising radiation where people are irradiated on purpose. The strategy for radiation protection is also different compared to that in other areas that have the zero-alternative as its ultimate goal, meaning that no human beings at all are exposed in these practices. The focus in x-ray diagnostics concerning radiation protection is justification and optimisation. Optimisation implies that the examination is performed in such a way that the radiation dose is as small as possible without jeopardising the diagnostic security.

X-ray diagnostics is a complex method where many technical parameters and methodology factors together are interacting in the determination of radiation dose and image quality. The optimisation process is not a simple and uncomplicated procedure, this difficulty is reflected in many international and national surveys showing a large spread of patient doses for one and the same type of examination. The concept diagnostic reference levels (DRL) has been introduced as a tool for reducing this wide distribution that is obviously indicating a lack of optimisation, and for cutting the highest radiation doses. In this presentation the concept for DRL and the experience gained in the Nordic countries with DRL are described.

What are DRL?

According to the European Council directive on medical exposures, 97/43/ Euratom (1), DRL are

Art. 2: dose levels in medical radiodiagnostic practices … for typical examinations for groups of standard-sized patients or standard phantoms for broadly defined types of equipment. These levels are expected not to be exceeded for standard procedures when good and normal practice regarding diagnostic and technical performance is applied.

Further the directive requires:

Art. 4 (2a): Member States shall promote the establishment and the use of diagnostic reference levels for radiodiagnostic examinations …and the availability of guidance for this purpose having regard to European diagnostic reference levels where available.

Art 6 (5): Member States shall ensure that appropriate local reviews are undertaken whenever diagnostic reference levels are consistently exceeded and that corrective actions are taken where appropriate.

So, the aim with DRL is in the first hand that practices having the highest radiation doses for the examination under consideration are identified and that measures are taken for reducing the radiation dose. Values below the DRL do not automatically imply that the procedure is optimised, but values exceeding the DRL are strongly indicating a non-optimised practice. A pragmatic approach for establishing DRL is to choose the 75 percentile of the doses from a national survey.

Criteria to be applied for DRL are that the measurements are simple with sufficient accuracy and that the quantity for DRL is correlated with the radiation risk, i.e. the effective dose or organ dose. Quantities meeting these criteria and which are used for conventional x-ray examinations are the skin dose (for single projections) or the dose-area-product, suitable also for complete and complex examinations. For mammography the average absorbed dose to glandular breast tissue is used and for computed tomography the dose-length-product and the average absorbed dose in the volume irradiated.

DRL in the Nordic countries

DRL is not a new concept in the Nordic countries. Already 1996 the *Nordic guidance levels for patient doses in diagnostic radiology* were published in the report series of the Nordic radiation protection authorities (2, 3) prepared by the Nordic working group on x-ray diagnostics. DRL (then called as "guidance levels") were defined for six conventional x-ray examinations, based on nation wide patient dose measurements in Norway, see Table 1. Three years later a study was performed in all the Nordic countries with the aim to compare the dose situation in the various countries and to see whether the DRL, based on the situation in Norway, are also applicable in countries outside Norway. The results have been presented at the European congress on radiology in 2000 (4).

An example for lumbar spine examinations is shown in figure 1. For each country the mean-, min- and max-values are shown, and the 25 and 75 percentile. The mean values for the various countries are equal with each other within a factor of two. The example for lumbar spine examinations is shown in figure 1. The number of x-ray stands is between 5 and 30 with 10 to 20 patients each. Although the distribution differs between the various countries the DRL chosen is acceptable, let be that it is somewhat high for Norway. The increased radiation doses in Norway compared with the earlier results depends most likely on the fact that now fluoroscopy is used for positioning before radiographic images are taken. Also for the other types of examinations the Nordic values for DRL are quite adequate, except for chest examinations where a lower value of 0,6 Gy • cm^2 has been shown to be more suited.

Table 1. Nordic guidance (=reference) levels for six diagnostic procedures.

Examination type	**Guidance levels**	
	DAP (Gy·cm²)	**ESD (mGy)***
Chest, PA and Lat	1	0,2 PA 0,5 Lat
Pelvis	4	5 AP
Lumbar spine	10	6 AP
Urography	20	
Barium meal	25	
Barium enema	50	

* Entrance surface dose including backscatter

Figure 2 shows the distribution of the radiation doses from barium enema examinations for all 89 units. The highest radiation dose differs from the lowest with a factor of 20, and similar spread is seen for the other examinations. Therefore we can draw the conclusion that there is a large potential for dose reductions. This has been verified in Sweden (5). The Swedish Radiation Protection Authority (SSI) has required for forty examination units, where the radiation doses were exceeding the DRL, that investigations about the cause for these values are performed and that dose reduction measures are to be taken. Thirty of these have reported the results to SSI: The dose reduction was between twenty and eighty-five percent with an average of forty-five percent. A positive finding was that many of the measures taken were simple and cheap and they should be possible to be realised even in hospitals with a stretched economy. However, it must be born in mind that the assessment and

Figure 1. Mean, min and max values of the dose-area-product for lumbar spine examinations in the five Nordic countries.

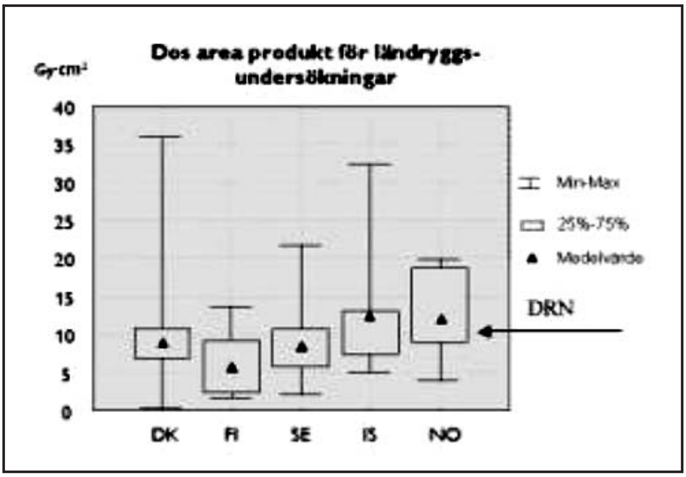
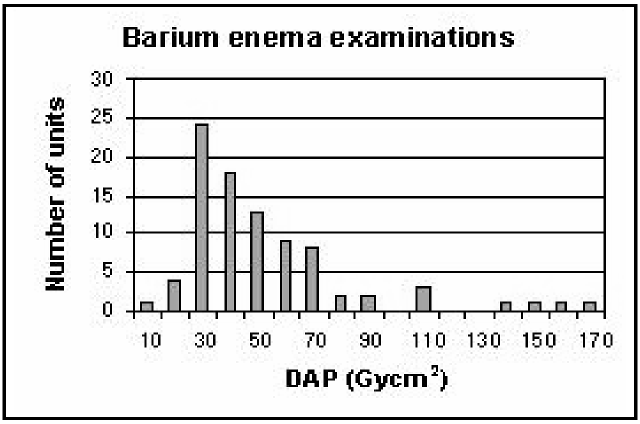


Figure 2. Dose-area-product for barium enema examinations (all Nordic countries).



follow-up of patient doses is entailing financial and personal resources, particularly in the initial phase.

Within the Nordic co-operation work has started for setting DRL also for interventional procedures (6) and computed tomography (7) but this work is not completed yet.

Implementation of DRL into national legistlation

The three Nordic member states in the EU have transposed the requirements in the medical exposure directive into national legislation. For five of the six conventional examinations the values proposed in (2) have been established as

DRL (except for chest examinations where Finland and Sweden have chosen 0,6 Gy • cm^2). The sixth examination in (2), barium meal, has become a rather uncommon examination because it has been replaced to a large extent by endoscopy and therefore not taken into the legislation.

For conventional examinations Finland has also established DRL for single views, with the quantity absorbed skin dose. For computed tomography European values (8) have been applied, slightly modified in the various countries based on own experiences. Denmark and Finland have chosen entrance surface exposure as the quantity for DRL in mammography, whereas in Sweden it is the average glandular dose in the breast, both for single views and for complete examinations.

The extent of measurements required, such as interval between measurements and number of patients, are somewhat different between the countries. Finland and Sweden are stating three years between measurements whereas Denmark has two years and for examinations comprising fluoroscopy one year. Ten patients are regarded as sufficient in Denmark and Finland; in Sweden the figure is at least twenty. Denmark is stating “patients of standard size”, in Finland the weight of the patients shall be between 55 and 85 kg with an average around 70 kg, in Sweden the patients preferably should weight between 60 and 80 kg with an average of 70±3 kg, but if this cannot be achieved the dose value for a 70 kg patient can be derived by interpolation.

Conclusion

Since more than six years work was performed in the Nordic countries with the concept diagnostic reference levels, to start with in a more limited and unsystematic way. The findings are good: just the fact that patient dose measurements are performed is drawing attention and interest to patient doses and the standardized method enables objective comparison between different hospitals. Remedial actions were taken for highest radiation doses found, often with little effort. Within the coming years thousands of patient dose measurements will be performed and compared with DRL. Based on the experiences gained up to now this will lead to considerable dose savings – probably a larger dose reduction then any other single action. A conservative figure would be a five percent dose reduction – in the Nordic countries this would lead to a dose saving of 800 manSv per year, an impressive figure in the context of radiation protection and in good balance with the inevitable costs entailed with the assessment of doses, with the investigations and with the remedial actions taken.

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Can the medical service afford to reduce radiation doses for x-ray examinations?

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Introduction

X-ray examinations are giving the largest contribution to the radiation burden of the public from artificial radiation sources. Mostly these examinations are justified; the benefit for the patient is normally exceeding the radiation risk by far. But they are not always optimised; there are indications that the radiation doses and hence the radiation risk frequently can be reduced without loss of the benefit for the patient. Shortage of resources is a frequently used argument brought forward by the medical service as a defence not to change existing conditions. In this presentation is accounted for which dose reducing measures have been taken, and their costs and effects are discussed.

Background

The Swedish Radiation Protection Authority, SSI, requested in 1999 that patient dose related data for a number of x-ray examinations were reported to SSI and these were compiled in a SSI report (1). The aim was to gain experience for planned regulations on diagnostic reference levels (DRL) and to gain information about the level of patient doses in Sweden. In total, data for around 8 000 patients have been reported, divided into 195 conventional, 148 computed tomography and 55 mammography examination stands. The spread of the radiation doses between the various examination stands for the same type of examination was varying with up to a factor of 15 between the highest and lowest. At the same time a realistic test of the concept with DRL was performed by applying those in the Nordic countries recommended levels for the conventional examinations – barium enema, chest, lumbar spine, pelvis and urography. When the patient dose at an examination stand was higher than the corresponding DRL SSI required investigation of the reason and that dose reducing measures were taken.

Table 1. Measures reported and the resulting dose reduction for the actual examination.

	Barium enema	Chest	Lumbar spine	Pelvis	Urography	Total
Number of examination stands (total)	35	45	40	30	45	195
- request for investigation	5	5	9	3	20	42
- no measures taken	-	3	3	-	6	12
- measures taken	5	2	6	3	14	30
Dose reduction for those that have taken measures (mean value)	60 %	55 %	35 %	45 %	35 %	40%
Min/Max dose reduction	20/85 %	50/65 %	20/60 %	25/55 %	15/60 %	15/85%

Measures for dose reduction

At 42 of the 195 examination stands for conventional x-ray examinations the radiation dose exceeded the Nordic reference levels. SSI has received reports on which measures have been taken at these stands and which effect these measures resulted in. Table 1 shows for how many examination stands measures have been taken and the total dose reduction achieved by these measures.

For 30 out of the 42 examination stands results have been reported from dose reducing measures. For the remaining 12 it was either declared that no possibility for dose reduction could be seen or that the equipment was to be replaced in the near future and nothing was done in between. For the five different x-ray examinations the average dose reduction was between 35 and 60% amongst those having taken measures. This can be seen as an indication that the introduction of diagnostic reference levels will provide a positive effect by reducing the patient dose. A prerequisite is that the costs for the measures taken are in reasonable proportion to the gain in radiation dose and that the measures do not render the diagnosis more difficult or even deteriorate it.

The majority of measures taken dealt with changes of the exposure factors (increased tube voltage, smaller field sizes), of the methodology (reduced number of images, reduced fluoroscopy time, use of compression). The latter was partly achieved by increased efforts on education of the staff. In seven cases the image receptor system has been changed and in four cases the whole equipment. An analysis of this material cannot give a conclusive answer to the question of how the result will be when DRL is introduced in a large scale, but it is giving hints on what could be expected. A mixture of what has been seen in the present study:

- *Measures requiring small resources*

  Education and training, change of the chemistry-film-system, increased developing time, increased filtration, increased tube voltage, reduced tube loading, reduced number of images, reduced fluoroscopy time and use of compression.
- *Measures requiring large resources*

  New equipment, direct digital image receptor, imaging plates.

But also that no measures are taken at all with the arguments:

- The diagnostic safety would be impaired, new equipment and/or accessories will be purchased in the near future.

At nineteen of the examination stands only low cost measures were taken. Some of them didn't cost anything, in certain cases there was even an economic benefit, e.g. when the number of images was reduced. Among these are measures such as reduced number of views, reduction of the fluoroscopy time or of the tube loading, increase of the tube voltage, increase of the filtration, smaller radiation fields, use of compression, change of film and chemistry in the film processor, increase of the developing time and education.

Even the change of the imaging system is a relatively low cost measure, taking into consideration its limited lifetime and the need for regular renewal. Even at the category where larger resources were needed, at six examination stands the analogue imaging system has been replaced by digital systems, the resources needed could be called moderate, and it is most likely that the replacement is part of a since long time ago planned development process towards digital x-ray images.

In less than 15 % of the cases a drastically measure such as replacement of the equipment has been taken. Most likely also this action is something laid down since long time ago in the investment plans for the hospital, let be speeded up because of the high patient doses. All this shows that it is realistic to expect large dose reductions with the introduction of diagnostic reference levels; the bad economy of the hospitals is certainly no excuse for not performing actions.

It cannot be concluded which influence the respective action has on the total dose reduction because mostly several different actions have been taken at the various examination stands. The effect is of course dependent on the starting point and on how much the respective parameter can be stressed. It is not possible on the basis of the present data to conclude which actions were most effective, but it can be stated that the potential for each of the actions mentioned is large.

The majority of the measures taken are about exposure parameters and examination methodology. This is an indication for that no systematic optimisation process has been conducted earlier at these stands. The good thing is that a large part of the high doses most likely can be reduced by simple and cheap actions, which increase the likelihood for them to be performed in the stretched economy of Swedish health care. In those cases where a drastic action such as replacement of the equipment is performed the exceeding of the DRL is most likely playing only a minor part, but a side effect could be that radiation protection issues will get more importance in the purchasing procedures of new equipment.

Conclusions

In the present study the concept of diagnostic reference levels has been used for the first time in a systematic way in Sweden. It has been shown that the patient doses could be reduced considerably by simple and cheap measures. Although the study had a limited scope it can be concluded that similar conditions exist when the concept with DRL is introduced on a large scale in the entire country. The examination stands with the highest doses will not only be identified but also measures will be taken to reduce the high patient doses because lack of resources is not a valid reason for refraining. It can be foreseen that patient doses are reduced considerably without jeopardising the diagnostic outcome.

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Clinical audit and quality systems - practical implementation in Finland

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Introduction

Clinical audit is a new concept of significant importance for the qaulity of radiological practices, introduced by the EC Medical Exposure Directive (MED, 97/43/EURATOM). By definition, clinical audit means "a systematic examination or review of medical radiological procedures which seeks to improve the quality and the outcome of patient care, through structured review whereby radiological practices, procedures, and results are examined against agreed standards for good medical radiological procedures, with modifications of the practices where indicated and the application of new standards if necessary". In its most profound meaning, being introduced in the medical exposure directive, clinical audit can be seen as a review of the success in implementing the justification and optimization principles, and therefore, it is to a large extent an issue of radiation safety for the patient. According to the directive, clinical audits shall be "carried out in accordance with national procedures".

For the last few years, parallel to the development of the MED in Europe, there has been a worldwide tendency to implement appropriate quality systems (QS) in the health care organizations, in accordance with the international quality standards (ISO 9000 series etc). Such quality systems have been applied for a long time and very widely by the industry. It is a strong belief that the development of quality systems for health care would result in equal benefits as trusted in industry, in terms of efficiency and safety of health care services. For radiological practices, the quality systems are expected to become a framework for improving the optimization of practices and for maintaining good radiation safety, as well as providing a mechanism to prevent mistakes and accidents.

In some countries, like the UK and The Netherlands, there are legal requirements to establish and maintain quality systems at certain type of radiological units. In some countries and some radiological units, quality systems have been developed voluntarily without any regulatory pressures. In some cases, the radiological units have acquired a certification of their quality system by a recognized certification body, or wanted to prove their competence by an accreditation by a recognized accreditation body (e.g., in some nuclear medicine laboratories). There has also been development to modify the general quality standards into more practical standards for health care applications. For example, the European Society for Therapeutic Radiology and Oncology (ESTRO) has published general philosophy and practical guidance on developing quality systems, specially tailored to the needs of radiotherapy, starting from ISO quality standards and published practice-specific quality assurance recommendations. Such practice-tailored recommendations or standards provide the health care organizations with an easy and straightforward access to building-up an appropriate quality system.

In the context of quality systems, confusion sometimes appears as to the meaning of “quality audits” versus “clinical audits”. Further, the difference of “clinical audit” from regulatory inspection needs to be clarified. In this presentation, these conceptual differences are discussed. The approach adopted in Finland for practical implementation of clinical audit is presented in detail.

Clinical audit and other quality audits

When an organization, e.g., a radiological department, builds-up a quality system, it may want that it conforms to a given QS standards (for example ISO 9001). To ensure this, the organization may apply for a certification of the QS, i.e., an independent evaluation by a certification body that the QS is in conformance with the selected quality standard. For such a certification, initially and regularly thereafter, a quality (system) audit by a certification body is introduced. The subsequent regular quality audits are needed to maintain the certification. The certification body has high expertice in quality standards and in general auditing procedures, but need not to employ health care professio-nals, because the purpose of quality (system) audits is only to check that the QS conforms to the requirements of the QS standard (a document which is non-specific to the radiological practice).

At least in radiotherapy, the term “quality audit” has been applied for a long time but with somewhat varying meanings. Generally, it can be understood as a “process audit” where the process is the dose delivery to the patient. In its most limited meaning, the quality audit has been no more than an independent check of the dose at the reference point in an external radiotherapy beam, e.g., by a mailed thermoluminescent dosimeter (TLD).

In a wider meaning, checking of a number of other dosimetric parameters of the beam, or simple testing of the accuracy of the treatment planning system, are also included. In its most comprehensive meaning, the quality audit in radiotherapy has been a review of the overall Quality Control and dosimetry programme of the treatment equipment. Even in this most comprehensive form, the quality audit has been limited to just one part of the radiotherapy process, the dose delivery to the patient. It has not addressed the total radiotherapy process that also includes, for example, practices for dose description, treatment preparation, treatment reporting and follow-up procedures.

Clinical audit is an assessment of radiological practices against pre-set criteria ("good practices") and should not be mixed with the above quality audit for QS certification. However, the quality audits such as the above examples for radiotherapy could be seen as important parts, but certainly not as the whole concept, of the clinical audit.

For a clinical audit, all grades of staff (all professionals) should be involved. A term "medical audit" has sometimes been used; it is equivalent to clinical audit, when only the medical staff (physicians, radiologists, oncologists etc) is involved.

Clinical audit and regulatory inspections

Clinical audit is in no way a regulatory action and should not be mixed with regulatory inspections. While regulatory inspections may sometimes include partly similar checking procedures or measurements as clinical audits, there are clear differences in the purpose of the procedures, in the criteria applied for the results, and in the authority of the persons performing the procedures. Regulatory authorities can carry out regulatory inspections for the verification of the fulfilment of legal requirements. Then the criteria are unambigious and always set in a law, decree or other regulations by the authorities. For clinical audit, the criteria are more freely set by the selected "good practice", For this, not a single choice but a number of good practices can be acceptable. Based on the results of the inspection, the inspector can impose orders or requirements to the licensee concerning the practice and the necessary actions for improvement. In case of clinical audit, the audit report with auditors' suggestions for improvements are given to the audited organization, while it is fully up to the audited organization to decide on the actions to be implemented. The auditors cannot impose any orders or requirements to the organization.

Practical implementation of clinical audit in Finland

Legislation

The basic requirement for the implementation of clinical audit has been set by a Radiation Law (1142/1998). The detailed requirements are given in the Decree on the medical use of radiation (423/2000).

According to the Decree, clinical audits shall be arranged to supplement in an appropriate way the self-evaluation of the practices. The goal shall be set so that all radiological practices would be audited for all essential parts at the minimum every five years.

The Decree states the following ten points which shall, among other things, be considered in clinical audits:

1 Definition of authority and responsibilities
2 Recommendation for referring a patient to other processes or departments
3 Practices and the information flow in justification process
4 Instructions and practices for delivery of medical exposure
5 Diagnostic and treatment equipment
6 Doses to patients and the outcome of diagnosis or treatment
7 Quality, storage and flow of information
8 Training of staff
9 Quality assurance procedures
10 Self-evaluation of practices

Organization

Representatives of professional societies, authorities and other key organizations have discussed the implementation of the requirements for clinical audit in Finland. This resulted in 2001 in the establishment of a special company, Qualisan Oy, to organize the practical implementation of clinical audits in Finland by developing and offering the necessary services. Qualisan is a joint-stock company established and supported by the major professional societies such as the Finnish Society of Radiologists and the Society of Radiographers in Finland. The company has organized a number of training courses for auditors, inviting volunteers from among the major professional groups (radiologists, oncologists, medical physicists, radiographers) to participate. About 40 auditors have been trained so far to undertake clinical audits through Qualisan, on request of the radiological units. The purpose is to train sufficient number of auditors for each health care district in Finland. The auditors are paid for their services while the audits are charged from the audited organizations.

A special advisory group for the development and follow-up of the clinical audit procedures, including a representative of a regulatory authority

(STUK), was established in close connection with Qualisan. Due to transparency and independence, this group is now planned to become a national steering committee fully independent of Qualisan.

Audit criteria

Qualisan has developed detailed criteria for the first round of audits, based on the ten points of interest specified in the Decree. In the future rounds of clinical audits, this criteria is planned to be supplemented by more practice-specific criteria, selected for each round of audits by separate considerations of the actual needs.

Conclusions

The organization and criteria for practical procedures of clinical audits in Finland have been developed, to provide the radiological units with easy access to implementing clinical audits. By the time of this writing, the training of auditors is still going on, while the first clinical audit was carried out in spring 2002 in a diagnostic radiology unit of a local hospital. The main problem so far seems to be the inadequate resources of the radiological units to build-up the necessary QS documentation to enable meaningful implementation of the audits. On the other hand, it has been stressed not to put too high a threshold for the level of QS before requesting clinical audit; even for a relatively "modest" quality system, the audit can provided the organization with important information and guiding for further development or completion of the system. A supplementary benefit of clinical audit is just to give additional pressure and guidance for the development of an appropriate QS.

Radiation doses from computed tomography

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Introduction

Computed tomography, CT, has experienced a tremendous development during the 30 years since its introduction. The first equipment was only able to produce images of the skull. The acquisition time as well as the reconstruction time was between 2 and 10 minutes. Even though the image quality at this time was poor, the new technique proved to be so useful in medicine that the inventors Allan M. Cormack and Godfrey N. Hounsfield were awarded with the Nobel Prize in medicine 1979.

Today the development of computers, mathematical algorithms, detectors and efficient x-ray tubes has made it possible to examine the whole body in less than 30 seconds and to produce 3D-images. This has lead to an increase of both the number of examinations and of the dos per CT-examination.

Justification and optimisation

Ct-examinations contribute with only 7% to the number of x-ray examinations but to more than 40% to the dose from medical exposure in

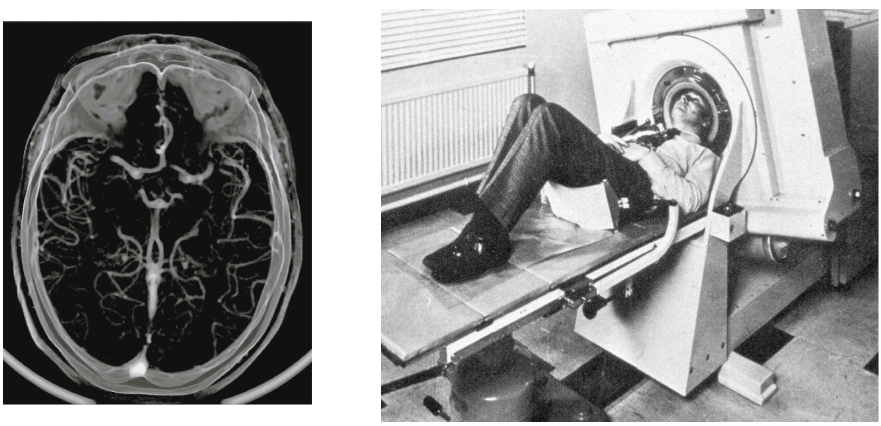

Figure 1.

Sweden (G Szendrö, 1995). This is not a problem if the examinations are justified, i.e. is doing more good than harm and if the examination is performed in an optimal way. Then it doesn't matter if the total use of radiation increases or decreases, what's important is whether the examination is useful for the patient or not. In a study 1999 (W Leitz, 2001) with the purpose to establish diagnostic reference levels (DRL) in Sweden, patient doses for 8000 examinations were measured, 3000 of which were CT-examinations. DRL is a by the Swedish Radiation Protection Authority (SSI) established level that should not be exceeded with good medical practice. It's not a limit, but an a investigation level, identifying potential bad practices. The DRL was chosen somewhat arbitrary as the third quartile of the dose distribution reported.

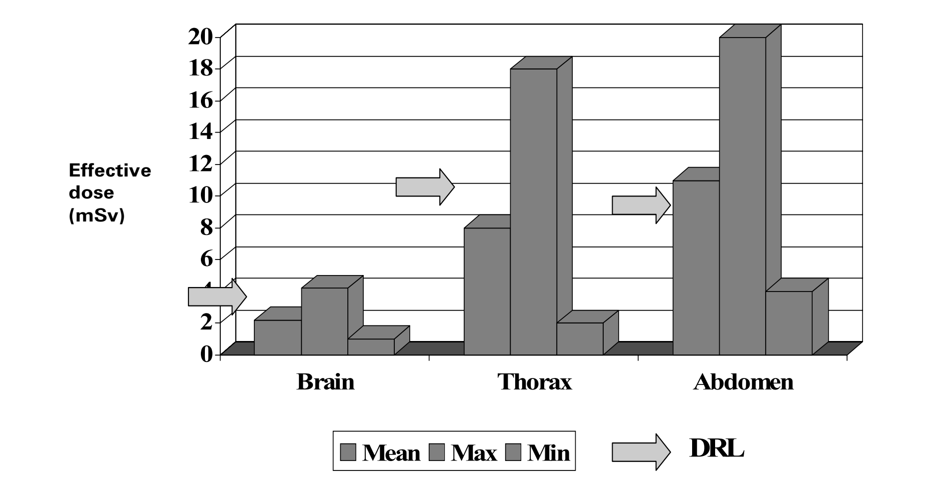

Diagram 1. dose distribution for brain-, thorax- and abdomen examinations.

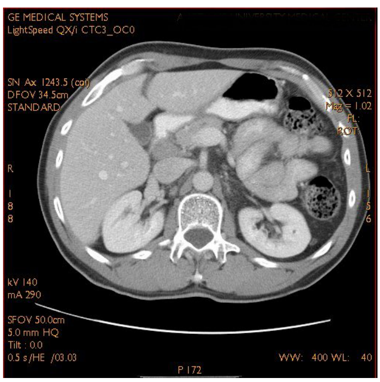

Figure 2. The image shows a slice of the abdomen with a region of interest marked as a black circle in the liver.

The study showed that effective dose could vary with a factor as high as 9 between different hospitals fore the same examination. For brain examinations the spread was from 1 to 4 mSv, for thorax from 2 to 18 mSv and for abdomen from 4 to 20 mSv. The wide spread in dose between different hospitals indicates that the examinations are not performed in an optimal way.

There are many factors influencing the patient dose, the tube voltage, the tube load i.e. mAs-setting, the slice thickness, the pitch factor i.e. table movement per rotation, and the scanned volume. All this factors are of course important fore the dose and imaging quality but in the rest of this paper I will concentrate on the importance of adjusting the tube load to the individual patient. This because we have noticed, that at many hospitals no concern is given to the individual patient. Instead a standard procedure is performed, i.e. the same settings are used for all patients.

Other factors that influence the noise in the image is the anatomy of the patient, or how thick the patient is, and the density of the examined organ. When going from the abdomen to the chest, the tube load (mAs-setting) must be reduced in order to achieve the same noise level. If not, the patient will receive a higher dose than necessary because of the higher transmission through the lung with its lower density. The problem with CT is that this will not be seen in the image let be that the images noise might be slightly reduced.

Optimisation of the tube load and image noise

By measuring the noise in a region of interest at a low tube load (mAs-setting) and then repeat the measurement in the same region for higher and higher tube loads the noise as a function of tube load can be plotted. The noise in the

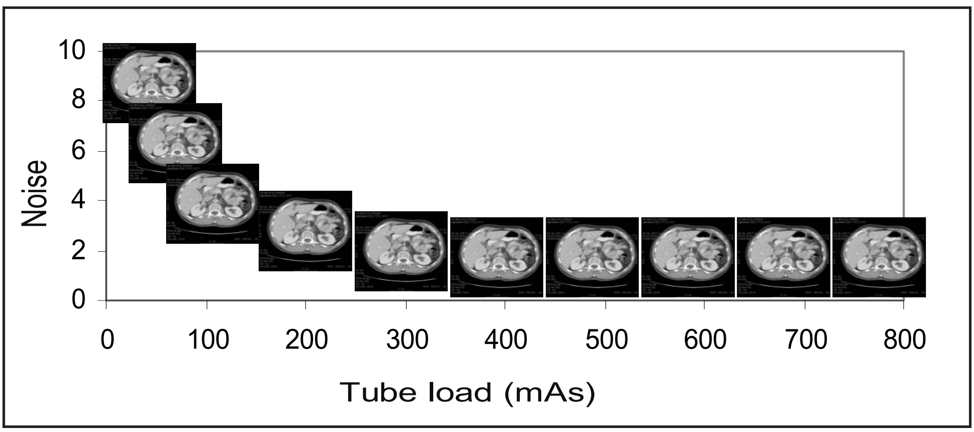

Diagram 2 . The image noise as a function of tube load.

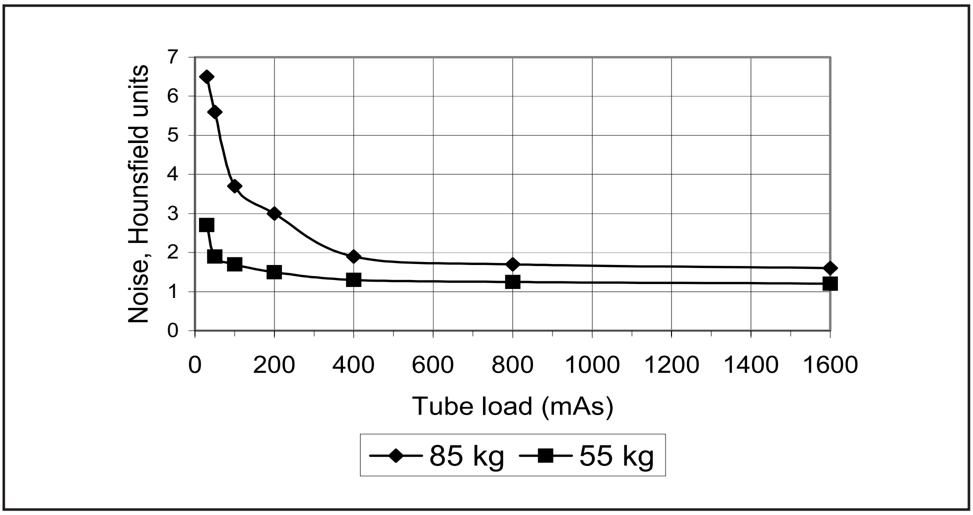

Diagram 3.

image will decreases rapidly in the beginning, but at a certain point this decrease will disappear and further increase in tube load will not improve the imaging quality. The point at witch the decrease will disappear is depending on the thickness of the patient and the density of the organ study.

If the image noise as a function of tube load for two different patients is compared, one that is weighting 85 kg and the other 55 kg, the potential of dose savings will be obviously. For the heavier patient the tube load should be approximately 300 mAs and for the smaller approximately 30 mAs. And unfortunately many hospitals will examine this two patients at a standard procedure with the same settings for both patient. This will lead to that the smaller patient will receive a dose 10 times higher than necessary.

Diagram 3 shows the image noise as a function of tube load for two different phantoms, one simulating a patient weighting 85 kg and the other one simulating a patient weighting 55 kg.

Conclusions

Bearing in mind that CT contributes with 40% to the collective dose from medical exposures, there is a large potential of dose savings. If all examinations are adjusted to the individual patient's anatomy the contribution to the collective dose would decrease without reducing the diagnostic security or the number of examinations.

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Nordic group on X-ray diagnostics: Intravascular brachytherapy – what it is and what the Nordic authorities demand

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Introduction

Intravascular brachytherapy is radiotherapy of the blood vessel to prevent it from re-narrowing after angioplasty. Angioplasty is a treatment for coronary artery disease where the artery that supplies the heart muscle with blood is blocked or narrowed. The radiotherapy is performed to decrease the growth of normal tissue and thus prevent the risk of restenosis, i.e. prevent it from re-narrowing. Intravascular brachytherapy after angioplasty has decreased the restenosis with more than 60%.

Stenosis in coronary artery

There are several different ways of treating stenosis in the coronary arteries. PTCA (percutaneous transluminal coronary angioplasty) is a method where a catheter with a small inflatable balloon on the end is guided through the blood vessels past the section of the artery that is narrowed. The balloon is the inflated several times to enlarge the opening in the artery and improve the blood flow. After the balloon inflation a stent, a thin metal grid, might be placed at the treatment area to keep the artery open. Up to 30 % of the patients get restenosis, often because of regrowth of normal tissue. The restenosis can appear as soon as after a couple of months after the treatment and will require

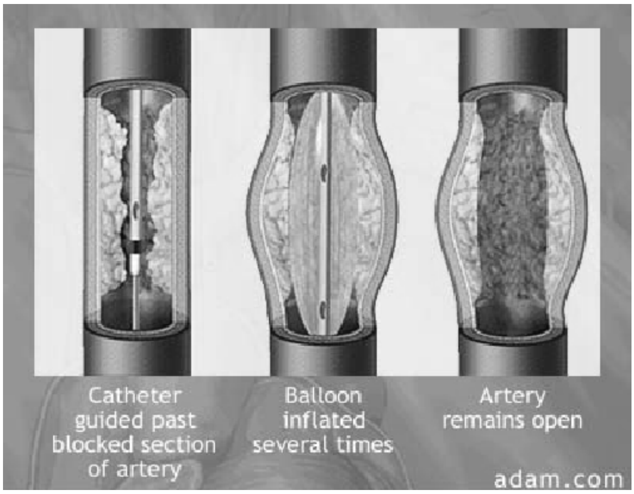

Figure 1. Inflatable balloon to enlarge the opening in the artery.

another treatment. A stenosis where the artery is totally blocked can result in a heart attack. In case of a multiple stenosis a by-pass surgery is made, i.e. a new blood vessel replaces the blocked one.

Intravascular brachytherapy

Intravascular brachytherapy is radiotherapy of the blood vessel and is made after the balloon inflation or after the stent implantation. The radiotherapy is given to decrease the growth of normal tissue and in this way decrease the risk of restenosis. The risk of restenosis has decreased from 30 % to 10-15 % for these patients.

The treatment is new and is expected to advance. The irradiation can be made with either beta or gamma emitters, for example ^{32}P, ^{90}Sr or ^{192}Ir. The treatment can be made in different ways, for example can the implanted stent contain a radioactive substance or the radioactive substance could be inside a balloon either as content or incorporated into the wall of the balloon. The treatment can also be made by sealed sources that are guided to or implanted at the treatment area. The activity in the radioactive source could be ten or so GBq, depending on type of treatment.

The most common technique is to guide a wire with the radioactive source on the end inside a catheter to the treatment area after balloon inflation. It is brought directly from the machine where it is stored in to the existent catheter in the patient, with minimal risk for contamination. The radiation of the blood vessel lasts for a couple of minutes and the typical

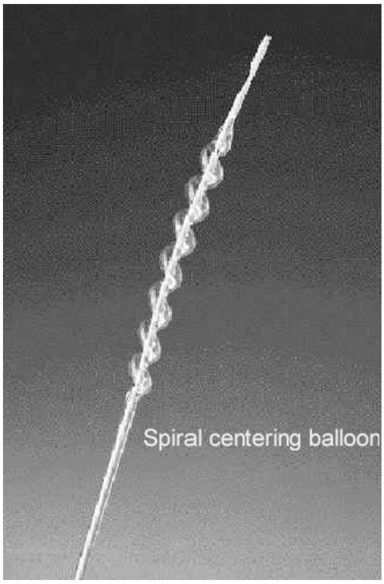

Figure 2. Balloon catheter as a spiral, where the wire with the radioactive source is centred in the middle.

absorbed dose to the artery is approximately 15 Gy. Considering the radiation safety for the staff, treatment with a beta emitter is preferable because of the small radiation dose to the staff. When the treatment is given with a gamma emitter, like ^{192}Ir, the radiation dose to the staff can be 10 to 20 microGy per treatment. The uncertainty in the dose planning can be as high as 60 %. In spite of this the positive effect of the treatment is obvious. Since the treatment is relatively new no long-time follow-up has been done

Nordic authorities

In the Nordic countries the treatment is given in six different hospitals, about as many plans to start the treatment. One problem is that the intravascular brachytherapy treatment procedure needs personnel from different disciplines and needs cooperation between cardiology, oncology and medical physics amongst others. Since so many different disciplines are involved the responsibility must clearly be assigned. The risk for incidents or accidents is higher since high activities are handled and the therapy is given in an environment not used to handling radioactive sources.

The authorities' demands on the different disciplines vary. All the Nordic authorities demand competence in cardiology and medical physics.

In some countries the presence of the medical physicist is needed during the treatment. The competence in oncology might not be needed or if it is needed the demand of presence vary. What kind of staff and how many that is supposed to take the education course offered by the company differs between the Nordic countries.

The handling of licences varies between the Nordic countries. In some countries a special licence for intravascular brachytherapy has to be applied for, while in other countries it is included in the existing licence for radiotherapy. Who signs the application vary also and the responsibility for the licence is ranging from the director to the cardiologist and medical physicist. No common quality assurance program for the Nordic countries has been developed yet, even though all countries have plans for emergencies and periodic controls. A Nordic guidance is to be written by the Nordic group on X-ray diagnostics.

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X-ray tube output based calculation of patient entrance surface dose: validation of the method

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Abstract
X-ray departments need methods to monitor the doses delivered to the patients in order to be able to compare their dose level to established reference levels. For this purpose, patient dose per radiograph is described in terms of the entrance surface dose (ESD) or dose-area product (DAP). The actual measurement is often made by using a DAP-meter or thermoluminescent dosimeters (TLD). The third possibility, the calculation of ESD from the examination technique factors, is likely to be a common method for x-ray departments that do not have the other methods at their disposal or for examinations where the dose may be too low to be measured by the other means (e.g. chest radiography). We have developed a program for the determination of ESD by the calculation method and analysed the accuracy that can be achieved by this indirect method.

The program calculates the ESD from the current time product, x-ray tube voltage, beam filtration and focus-to-skin distance (FSD). Additionally, for calibrating the dose calculation method and thereby improving the accuracy of the calculation, the x-ray tube output should be measured for at least one x-ray tube voltage value in each x-ray unit. The aim of the present work is to point out the restrictions of the method and details of its practical application. Our first experiences from the use of the method will be summarised.

Introduction
The diagnostic reference levels (DRL) were introduced in the EU-legislation by the Medical Exposure Directive (MED) [1]. Therefore, in order to be able to compare their dose levels to established DRLs, x-ray departments need methods for monitoring patient doses.

A working party of professional organisations prepared the UK national protocol for patient dose measurements in diagnostic radiology [2] using the experience from the survey organised in UK in the 1980s [3]. In that protocol, the entrance surface dose (ESD) was recommended to be used as the quantity to be measured for individual radiographs [4]; the measurement was based on the use of TLDs. After that time, the use of dose-area product (DAP) meters and methods for calculating of ESD have also become common. These methods result to the determination of the entrance air kerma without backscatter from the phantom (to obtain the ESD, the DAP method requires also the determination of the x-ray beam size at the patient entrance plane). In order to get compliance with TLD measurements, the results must then be multiplied by the backscatter factor (BSF).

The aim of the present work is to reduce problems in using the ESD calculation method. We intend to simplify the calculation by publishing a computer program which will allow the x-ray technicians to perform the ESD calculations at their departments. Further analysis of the doses and a comparison to reference levels may also be made using the program. The program is being developed at the Radiation Metrology Laboratory of STUK (Radiation and Nuclear Safety Authority, Finland).

Calculation of ESD

The ESD is defined as the dose absorbed in air at the surface of the patient on the central axis of the radiation beam, and includes the radiation backscattered from the patient (although other definitions exist as well, involving the dose in tissue instead of air and/or exclusion of backscattered radiation). The ESD can be calculated if the radiation yield of the x-ray tube, the current-time product (Q), the focus-to-skin distance (FSD) and the backscatter factor are known [5]. Then

$$ESD = Y_{100} \cdot (100\,cm\,/\,FSD)^2 \cdot Q \cdot BSF\,. \qquad (1)$$

Here, Y_{100} is the output of the x-ray tube (mGy/mAs) measured at 100 cm distance from the x-ray tube focal point. For a given x-ray system, Y_{100} depends mainly on the x-ray tube voltage and filtration. Variability of Y_{100} between different x-ray systems exist, and can be addressed to several factors, including at least the accuracy of indicated values (kV, mAs, filtration), the x-ray tube voltage wave form, the x-ray tube anode angle, the smoothness of the anode surface, various attenuating materials in the primary beam and differences in the amount of off-focal radiation in various units.

Our ESD calculation program uses the method described in Eq (1). The program is written in Borland Delphi Object Pascal code and can be run under

Windows 9X/NT/2000/XP operating systems. An essential part of the program is the automatic calculation of the x-ray tube radiation yield, which was included to minimise the extra work needed in using of the method in x-ray departments. The calculation of Y_{100} uses the method of generating computed x-ray spectra by the theory of Birch and Marshall[6]. In addition to this, an empirical curve depicting the x-ray tube output with a fixed amount of filtration is used. The actual beam filtration is taken into account by modifying the computed x-ray spectrum by the difference of the actual filtration to the reference filtration, and calculating the change to the reference radiation yield that the modified filtration would cause. This calculation results to the expected radiation output of the x-ray tube, at the x-ray tube voltage and filtration that are being used. All the factors mentioned above, affecting x-ray tube output, cannot be modelled in the calculation. The resulting inaccuracy of the dose calculation can then be taken care of by measuring the x-ray tube output at one or several high-voltage settings and/or filter choices, calculating a calibration factor (the quotient of the measured and expected doses) and using this figure for all high-voltages and filters. The resulting calibration factor can also be used as quality control tool: a relatively large (> 30 %) difference between the measured and the expected tube output may indicate problems in the adjustment of the x-ray system.

The selection of backscatter factors approximated in the program was based on the technique factors suggested in [9] and typical radiation field sizes and FSDs used in the examinations. BSFs were taken from data calculated with the Monte Carlo method [10,11] and measurements [2].

Uncertainties in the calculated ESD

The standard error between the measured and calculated output was approximately 16 % when no individual calibration data for the x-ray tube was applied: this corresponds to the observed variability in the output of diagnostic x-ray tubes at a fixed filtration. However, if the measured x-ray tube output is used for calibrating the calculation, the agreement between the measurements and calculated doses gets better. With a single calibration point (a single kV-setting used for normalisation) for each of the x-ray units considered the standard error between measured and calculated values was 5 % in the whole range of kVs and filters that were measured. An example of this is shown in Figure 1, which shows the improved agreement between calculated and measured data when the calculated data have been calibrated.

Regardless of the actual materials in the x-ray beam, the total filtration of an x-ray tube is normally expressed as a thickness of aluminium. This equivalent aluminium filtration results to a similar x-ray spectrum shape (and HVL) as the original filtration, but the intensity of x-radiation may be different when the materials and their thicknesses are not the same. Typically, the x-ray beam path of diagnostic x-ray tubes includes glass and insulating oil in addition to the actual aluminum filter. In some x-ray tubes, the window is made of beryllium: then, the radiation intensity will be higher than in a glass-window tube. According to our calculations, the effect could be of the order of 20% and is only weakly dependent on the x-ray tube voltage.

Our reference radiation yield curve does not consider the effect of the target angle either, although this factor may have a notable impact to radiation output and spectrum shape[7]. The difference between x-ray tubes with a 10° and 22° target angle may be of the order of 20–40 % depending on the smoothness of the target. The dependence on the x-ray tube voltage seems to be weak.

A further factor that is not explicitly considered in our reference radiation yield curve is off-focal radiation (radiation generated outside of the focal spot). Its amount may vary with the x-ray tube and collimator types, and be also dependent on the x-ray field size. Birch *et al*. [7] stated that the

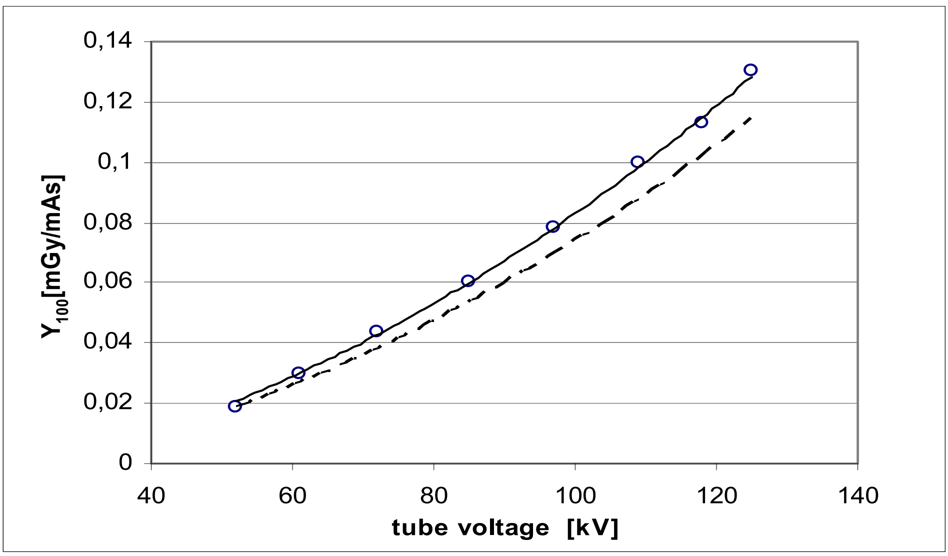

Figure 1. An example of the calculated and measured (circles) radiation yield of x-ray tube, Y_{100}. The calculated curve is presented with (solid line) and without (dash line) calibration. One radiation yield measured at 85 kV has been used for the calibration of the program. The total filtration of the x-ray tube is 3 mm aluminium.

contribution of off-focal radiation to the total output is between five and ten percent. According to Weaver *et al*. [8] the contribution could be as high as 25 %.

Our calculation method does not explicitly consider factors relating to the actual filter materials, the angle or smoothness of the x-ray tube target, or off-focal radiation. These factors are included by the calibration method explained above.

Of course, the accuracy of the indicated values (kV, mAs, filtration) is essential for the proper functioning of the ESD-program. Therefore, the user should ascertain that the x-ray tube voltage and the filtration are as indicated. These can be checked by penetrameter and HVL measurements during quality control measurements, for example. One should also ascertain that the x-ray tube output is linear with respect to tube charge (mAs) for the whole mAs-range used clinically. Often, tube output may be lower than expected for very short exposures.

Inaccuracies of other data result to uncertainties, too. For example, an error of 1 cm in the measurement of FSD causes a 3% error in the ESD at FSDs of the order of 75 cm and the BSFs may vary by a few percent from patient to patient. There is also variations in backscatter factors due to different tube voltages and field sizes used in the examinations. The maximum difference between the BSFs derived from the references mentioned above is approximately 5%.

Comparison between measured (TLD, DAP) and calculated ESD

The ESDs calculated with the program were compared with patient doses measured with TLDs and DAP meters in five different x-ray departments. The total number of examinations was 148. The type of the examinations, current-time products, tube voltages, total filtrations, FSDs and the patients' height and weight were recorded to enable the calculation of ESDs.

Table 1. The averages and standard deviations of the ratios of the calculated ESDs, the ESDs measured with DAP meter and the doses measured with TLDs. The ESD-calculation has been made with one x-ray tube radiation yield calibration value for each x-ray tube.

Type of examination	ESD_{calc}/ESD_{TLD}	Std (%)	ESD_{calc}/ESD_{DAP}	Std (%)
Abdomen AP	1,03	8,3	0,94	8,3
Lumbar spine AP	1,03	6,7	1,04	6,2
Lumbar spine LAT	1,01	8,9	0,97	6,0
Chest LAT	0,99	16,6	1,01	8,2
Chest PA	1,14	11,3	1,07	12,7
Pelvis AP	1,00	9,4	0,90	6,1

The average ratios of the calculated and measured ESDs are presented in table 1. The values show that there is good agreement between the calculated ESDs and those measured with TLDs or DAP meters. The slightly high ratio of calculated and measured (TLD) ESD in chest radiography in table 1 is partly caused by the low doses in those examinations: the doses may have been too low to be measured accurately with TLDs. In the other than chest examinations it can be approximated that the inaccuracy of calculated ESD is between about 13% and 20% when compared to TLD measurements.

Discussion

Several things require special attention in using the ESD calculation method. For example, errors in measurements of FSD and current-time product have great impact on total ESD. The theoretical calculation of the radiation yield decreases possible errors arising from manually using tube output curves by inexperienced personnel. Anyway, the user must verify that the radiation yield of the x-ray tube is normal and calibrate the program before using it. By using the theoretical radiation yield without calibration the uncertainty of the yield is about 30%. Adding one calibration value in tube information decreases the uncertainty to less than 10%. The backscatter factors used by the program have been chosen so that they correspond to typical tube voltages, total filtrations and field sizes in diagnostic radiography.

If calculational methods or DAP measurements become the main method of dose assessment, it would be useful to revise the reference doses to be expressed in terms of the entrance air kerma free-in-air (excluding backscattered radiation). This would help by removing one of the uncertain factors in dose assessment.

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Radiation dose measurement of paediatric patients in Estonia

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Introduction

According to the Medical Exposure Directive (97/43/Euratom) the radiation doses to patients should be measured in every hospital and doses should be compared to the reference doses established by the competent authorities. Special attention should be paid to the paediatric x-ray examinations, because the paediatric patients are more radiosensitive than adult patients. The requirement of measurements of radiation dose to patients is not yet included in the Estonian radiation act, but the purpose to join the European Communities makes the quality control in radiology very actual in Estonia. The necessity exists to introduce suitable measurement methods in the X-ray departments of Estonian hospitals for establishing feedback system for radiologists, radiographers and medical physicists in optimising the radiation burden of patients and image quality.

Materials and methods

Patient doses in various paediatric radiographic examinations (of pelvis, chest, spine and head) were measured in three X-ray departments. Two of these (H1 and H2) were located in children's hospitals, and the third (H3) was in a general hospital. Data were collected for more than 400 radiographs of children referred to the X-ray departments during a period in 2000 and 2001.

Entrance surface dose (ESD) to patients was estimated from DAP measurements and by calculations using the data of x-ray tube output and examination techniques, expressed respectively as ESD_{DAP} and ESD_{Y}.

The dose-area-product (DAP) was measured with DAP meter and entrance surface dose (ESD) was calculated by using examination techniques and DAP data by the formula:

$$ESD_{DAP} = \frac{\lambda \times DAP_{read} \times FFD^2 \times BSF}{A \times FSD^2} \qquad (1)$$

where λ is the calibration coefficient (dependent on tube potential) for DAP meter, DAP_{read} is the reading of DAP meter, *BSF* is the backscatter factor (dependent on radiation field size and half value layer (*HVL*)), *A* is the field size measured on the film from the patient examination, *FSD* is the focus-to-skin distance, *FFD* is the focus-to-film distance used in this examination.

The entrance surface dose ESD_Y (including backscatter) was calculated by using the formula:

$$ESD_Y = \frac{Y_{100} \times MAS \times FFD_{100}{}^2 \times BSF}{FSD^2} \qquad (2)$$

where Y_{100} (mGy/mAs) is the X-ray tube yield (dependent on tube potential and filtration) at a standard distance of 100 cm from the tube focus, *MAS* (mA×s) is the product of tube current (mA) and exposure time (s), FFD_{100} is the film to focus distance (100 cm) used in the tube yield measurements.

For dose-area product measurements DAP meter models 840A (Gammex RMI, USA) and VacuDAP 2004 Type 70157 (VacuTec Meßtechnik GmbH, Germany) were used in different hospitals. DAP meters were calibrated *in situ* (i.e. DAP meter was connected to the collimator) against the radiation monitor MDH 1015 at various tube potentials. The MDH 1015 was calibrated in the secondary standard laboratory at the STUK, the calibration being traceable to the National Physical Laboratory (NPL) primary standard.

The uncertainty in the DAP and ESD measurements was assessed. When measuring patient doses, a value of dose-area product is calculated from the dose-area product meter reading DAP_{read}, multiplying it with the calibration coefficient λ at the relevant tube potential. Therefore, the relative standard uncertainty of *DAP* is calculated by using the relative uncertainties of both quantities:

$$\delta u(DAP) = \sqrt{\delta u(DAP_{read})^2 + \delta u(\lambda)^2} \;. \qquad (3)$$

The expanded relative uncertainty at k = 2 (k means coverage factor and k = 2 means 95% confidence interval) in the DAP measurements was estimated as ±10 % for DAP results about 100 mGy·cm^2 (and at high tube potentials) and ±17 % for results about 10 mGy·cm^2 (at low tube potentials, with added filtration). These estimates of uncertainty in the *DAP* measurements are still well below the maximum acceptable combined standard uncertainty (k = 2) of ± 25 % stipulated in the international standard IEC 60580 for dose area product meters.

The relative standard uncertainties in ESD_{DAP} and ESD_{Y} measurements were assessed respectively by the Formulas 1 and 2. The expanded uncertainty (k = 2) in entrance dose ESD_{DAP} measurement at different tube potentials was assessed as minimum as ± 17 % (at 120 kV and at DAP of 100 mGy·cm^2) and as maximum as ± 23 % (at 50 kV and at DAP of 10 mGy·cm^2). For entrance dose ESD_{Y} at all values the uncertainty was assessed as ± 11 %.

Table 1. Measured and calculated dose quantities for different age groups at H1.

Radiographic examination (projection)	Age group (years)	Number of patients	Mean DAP (mGy·cm^2)	Mean ESD (mGy)
Head Lat	1 - 4	16	15	0.49
	5 - 9	25	18	0.57
	10 -16	12	24	0.68
Head AP/PA	1 - 4	3	104	1.1
	5 - 9	21	134	1.4
	10 -16	25	136	1.5
Pelvis AP	<1	32	8	0.067
	1 - 4	5	90	0.48
	5 - 9	3	137	0.45
	10 -16	2	307	0.69
Chest AP/PA	<1	18	8	0.050
	1 - 4	45	13	0.062
	5 - 9	41	32	0.097
	10 -16	42	43	0.096
Spine AP	5 - 9	5	197	0.68
	10 -16	12	352	1.2
Spine Lat	1 - 4	1	200	0.93
	5 - 9	4	249	0.91
	10 -16	7	601	2.4

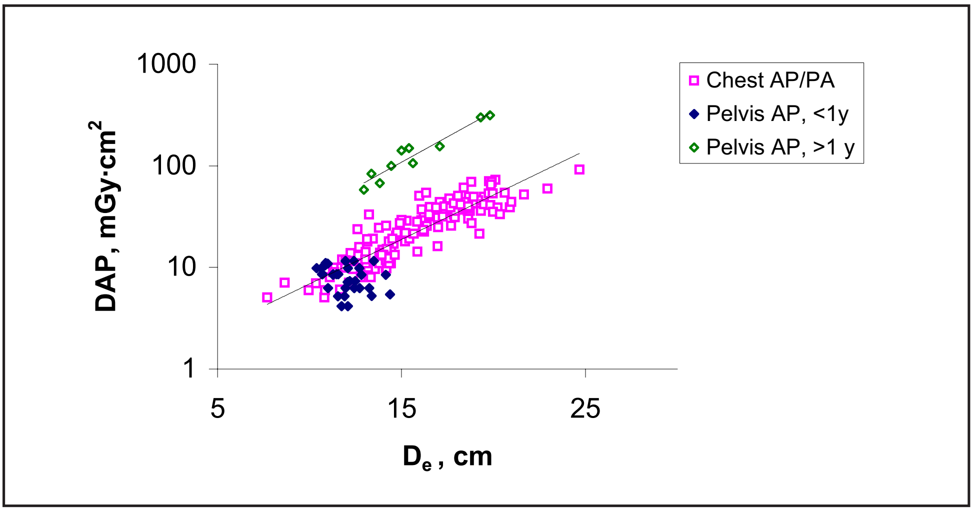

Figure 1. Correlation between dose-area product (DAP) and equivalent diameter (D_e) of the patient in chest and pelvis <1 year (without grid) and pelvis >1 year (with grid) examinations.

Results

Dose-area products (DAP) and entrance skin doses (calculated by DAPs) for six common head and trunk X-ray examinations of different paediatric age groups are presented in Table 1. Doses for the same projections in the same age groups differed usually up to 5 times.

Figure 1 shows the variation of log(DAP) values by equivalent diameter of the patient D_e (or ECD [2], calculated by the formula that includes weight and height of the patient, thus taking the average density into consideration) for examinations of chest and pelvis. The linear correlation is significant for pelvis examination of the age group >1 y and chest examinations.

In the age group of infants and occasionally in the next age group (1-4y) for pelvis AP examinations no grid was used. Comparison of DAP values derived from fits of the data for log (DAP) *versus* equivalent cylindrical diameter D_e (Figure 1) showed that DAPs for pelvis radiographs without a grid were 10 % of those when a grid was used for a 14 cm patient.

In Figures 2 to 4 the mean DAPs and ESDs from the hospitals H1, H2 and H3 are compared against the levels of the best practice in paediatric radiology based on results of UK survey by Cook et al. [1], against the Finnish survey by Servomaa et al. [2] and against the CEC reference dose levels for 5-year-old patients [3]. The error bars show the actual data range at H1.

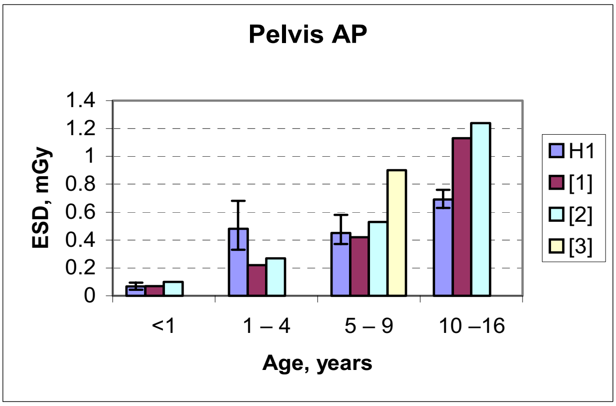

Figure 2. Mean ESD in pelvis AP examinations at H1 compared with the results from other surveys by Cook et al. [1], Servomaa et al. [2] and EC [3].

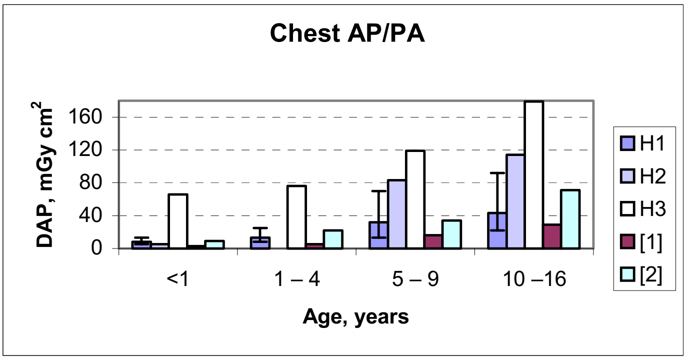

Figure 3. Mean DAP in chest AP/PA examinations at H1, H2 and H3 compared with the results from other surveys by Cook et al. [1] and Servomaa et al. [2].

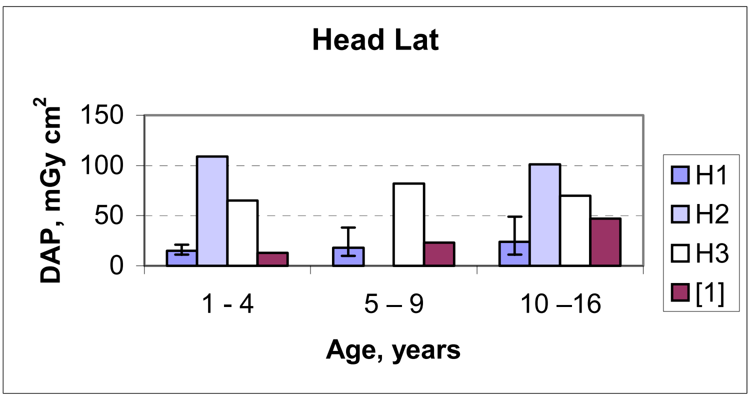

Figure 4. Mean DAP in head LAT examinations at H1, H2 and H3 compared with the results from the surveys by Cook et al. [1].

Conclusions

The measurements reported in this study establish the basis for dose measurements as part of the radiological quality assurance in Estonia. These preliminary results are just indicative, but are comparable with the results of other European surveys. The patient dose measurements are to be continued to establish the national dose reference levels.

It was shown that *ESD* can be derived from DAP meter readings to an 95% uncertainty of ± (17-23) %. The lowest value of *ESD* measurable by this method was primarily limited by the resolution of the DAP meter (1 mGy cm^2). However, because of the relatively large uncertainty associated with this method, *ESD* can be estimated also by the tube yield and examination parameters, with the expanded uncertainty of ±11 %.

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Posters

7a Nordic radiation protection co-operation: Report from activities in the task group x-ray diagnostics

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Abstract

For more than 25 years scientists and executive regulators from the five Nordic countries have collaborated within the framework of Nordic co-operation in the field of radiation protection in X-ray diagnostics. The task group "X-ray diagnostics" has published a number of recommendations, reports and scientific papers, and the work in the group is summarised in this paper.

History

In the Nordic countries, a formal and practical co-operation between the radiation protection authorities has existed for decades. The scope is to exchange information of mutual interest, discuss regulatory questions, and for appropriate matters produce common reports and recommendations. The directors general from the Nordic radiation protection authorities meet regularly. From this forum dedicated task groups are appointed to solve various matters. Some working groups have had a more dedicated commission and were dissolved after finishing their task. On the other hand, the task group "X-ray diagnostics" has existed for more than 25 years, dealing both with regulatory questions, and scientific activities. The "chief meeting" nominates the chairman in this group, and one contact person from each country are selected more informally, while other contributors participate on a short notice depending on the topics.

Many former chairmen and central co-workers such as Jon Flatby, Sten Grapengiesser, Jonas Karlberg, Tord Walderhaug, Olli Ojala, Gunnar Saxebøl and Ole Hjardemaal, are remembered for their excellent contributions to radiation protection in diagnostic radiology, not to discredit the work of many of the other members throughout the years.

Report series and advisory activities

A certain report series was designated to Nordic co-operation, popularly called the "Flag-book" reflecting the five flags from each of the Nordic countries on the front page. The most basic publication in this series was brought out in 1976, issuing an implementation of the ICRP recommendations in the Nordic countries[1)]. The task group X-ray diagnostics have published one report in this series, dedicated protection of the foetus [2)].

Our working group suggested another report series, as a supplement. The aim was to have a report series with less administration and publishing expenses. The system is coordinated from the Norwegian Radiation Protection Authority (NRPA), who prints the covers, front pages and colophons. The editors of each report supply each country with a master copy of the content. The report series is dedicated ISSN 0804-5038, and reports may be ordered from each of the national radiation protection authorities. The task group X-ray diagnostics has published seven reports

Table 1. Reports published in the series "Report on Nordic radiation protection co-operation".

No 1	«Om mammografi» (svensk tekst, 1990) «Mammography» (english text, 1994)
No 2	«Shielding of gonads» (english text, 1994)
No 3	«Kvalitetskontrol og Tilsyn med Medisinsk Røntgendiagnostisk Udstyr. En Oversigt.» (dansk tekst, 1994)
No 4	«Glandular tissue dose in film-screen mammography» (english text, 1995)
No 5	«Nordic guidance levels for patient doses in diagnostic radiology» (english text, 1996)
No 6	«Radiografutdanningen i Norden-innhold av realfag og strålehygiene» (norsk tekst, 1996)
No 7	«A quality Control Programme for Radiodiagnostic Equipment : Acceptance tests» (english text, 1999)

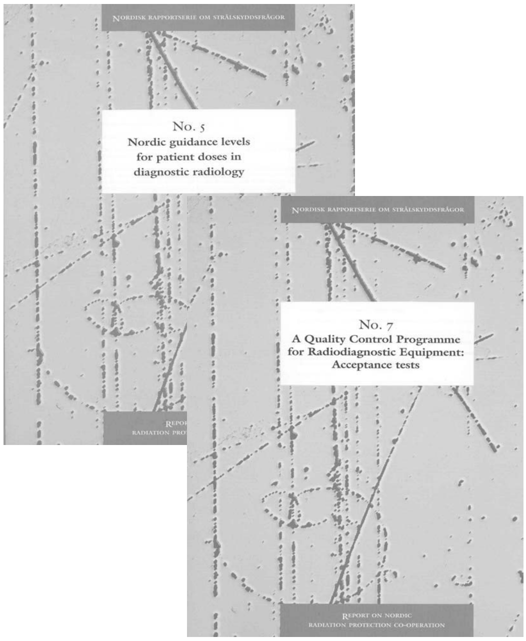

Figure 1. Layout of the series "Report on Nordic radiation protection co-operation".

so far [3 – 9]. The titles of the reports reflect the scope of the task group, dealing with radiation protection of the patient, education and competence of the staff, quality control of radiological equipment, and the establishment of diagnostic reference levels. A list of published reports in the series is shown in the table below, and the layout of the reports can be found in figure 1.

Scientific activities

The task group X-ray diagnostics has also been active in the international radiological society, and published a set of scientific papers, with emphasis on the establishment of reference doses in diagnostic radiology. As dose measure for conventional radiography and fluoroscopy, the Nordic countries have selected the kerma-area product (KAP), which is measured with a plane-parallel transmission ionisation chambers intercepting the entire X-ray beam during examinations[10, 11]. This quantity is also applied for interventional radiology, even though the value of entrance skin dose should always be recognised[12]. By pooling the data from national surveys in each of the five countries, the statistical power is increased, and we have a better foundation for the establishment of the reference

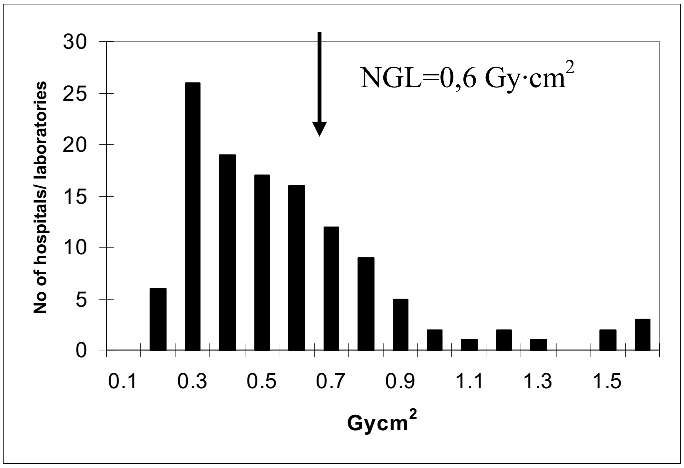

Figure 2. Based on measurements in a selection of hospitals in all the five Nordic countries, the total distribution of the kerma-area product (Gy · cm^2) are shown for examination of chest (Eur. Radiol. 10, 1988 – 92 (2000))[11).

dose values. An example is shown in figure 2.

A joint survey was also arranged in 2000, considering both patient dose and image quality in computed tomography (CT)[12). The weighted CT doseindex, the dose-length product, as well as the effective dose, were used as patient dose parameters. The results from this pilot-study serve as input to the establishment of reference doses in CT.

Future tasks

In the near future the task group X-ray diagnostics will focus on the following topics

- Implementation of reference levels in conventional radiology, a process dependent of the national regulations and the availability of measuring equipment and local competence
- Establishment of reference levels in CT and in interventional radiology (one or two procedures) (2002). A need for guidance and teaching in the hospitals about the understanding of the specific dose quantities introduced with CT is recognised.
- A pilot study for optimisation of CT protocols for paediatric patients (2003)
- A recommendation for the management of endovascular brachy therapy will be published as No 8 in the Nordic report series (2003) in collaboration with a recently established Nordic working group on dosimetry and radiotherapy.

Conclusion

Certainly, the development in diagnostic radiology challenges the radiological protection society. The widespread use of X-rays in medicine means that such examinations represent by far the largest man-made source of population exposure to ionising radiation. The principal concern in radiological protection is the reduction of unnecessary exposures by requiring adequate clinical justification of an investigation and by optimisation of the patient investigations to give minimal dose for sufficient diagnostic information. The rapid technological development, the digitalisation of the radiological departments and new modalities introduce the need for constantly new efforts, and updating of our knowledge. We have for long been aware of the increasing contribution from computed tomography (CT) to the collective effective dose. The reduction of exposures by requiring optimisation of CT procedures is therefore of principal concern in radiological protection. This is especially the case for paediatric patients. We have seen that some interventional procedures may give skin burns, and need special attention. The implementation of reference levels is one of several tools to meet those challenges.

We also need to manage the propagation of new modalities such as endovascular brachytherapy. What are the requirements of dosimetry, quality control procedures and safety in source handling? Which categories of physical and medical professions should be available to get license for conducting such laboratories? Nordic co-operation is valuable to discuss such matters, and sometimes we come to a mutual point of view that probably will have a wider impact in the society compared to each countries separate policy.

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7 b Measurement of staff doses in interventional procedures using LiF TL-detectors and a special diode dosemeter

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Introduction

Personal doses vary considerably among the operational staff in cardiac examination procedures and interventions. Correspondingly, the fluoroscopy time may vary from a few minutes to more than 1 hour. Large personal doses in the range of 35 - 41 mSv were registered to some interventional radiologists and cardiologists in 2001 in Finland. Measurement of doses per intervention from hands, shoulders and ankles (parts of the body outside the lead apron) are therefore of interest, but small and highly sensitive dosemeters have not been commonly available for reliable and practical measurements. Thermoluminescent detectors of LiF:MgCu,P are sensitive enough, but residual luminescence from previous uses of detectors and the dose collected from natural background radiation between preparation for use and readout may be significant sources of uncertainty. The company Unfors Instrumens AB (Billdal, Sweden) has designed a light weight diode dosemeter with long cables between detectors and the electronic part for simultaneous measurement of doses from various parts of the body. The aim of this work is to report doses of the operating staff per a cardiac examination or intervention and study the capabilities of the diode dosemeter in staff dose measurements.

Methods

Measurement quantities

The manufacturer has calibrated the diode dosemeter Unfors EDD-30 to measure the personal dose equivalent $H_p(10)$ and $H_p(0,07)$. Because those quantities are not ideal for extremity dose measurements and to achieve comparability between the results of the diode and the TL dosemeters, air kerma was used as a measuring quantity in this study. For an ideal dosemeter, air kerma can be considered as an estimate of the (extremity) dose equivalent H_p (0,07) within 3%. For convenience, the terms ´*dose*' and ´*dose rate*' are used instead of '*air kerma*' and '*air kerma rate*' when referring to the measurements of personal staff doses.

Tests of diode dosemeter in laboratory conditions

The response of the diode dosemeter EDD-30 for air kerma relative to the X-ray quality (kV and filtration) and relative to air kerma rate was studied. To produce low air kerma rates irradiation distances up to 5 m and water slab beam modifiers were used. Radcal 9015 dosemeter with the 180 and 1800 cm^3 ionisation chambers was used as a reference dosemeter in the these tests.

Staff doses measured with TL dosemeters

The ultra sensitive LiF:Mg,Cu,P TL dosemeters were obtained from the company Doseco Oy (Jyväskylä, Finland). TL- detectors were placed on the shoulder, ankle and hand of the cardiologist and radiographer before the start of the examination and intervention (Satakunta Central Hospital, Pori, Finland). After the procedure each detector was disconnected from the staff for readout later on. Each detector was thus used only during one procedure, and altogether 200 detectors were needed for the 29 CA (cardioangiography), 12 PTCA (percutaneous transluminal coronary angioplasty), and 4 pacemaker procedures included in the study. The dose-area product (DAP) was also recorded from each procedure to illustrate the patient doses of procedures. The TL detectors were calibrated beforehand for air kerma and by X-ray quality of 80 kV and of 3 mm Al.

Comparison of diode and TL dosemeters for measurement of the staff dose

One diode dosemeter including electronic device and one diode detector was available for the comparison within TL dosemeters. The diode detector was fixed close to its reference TL dosemeter either on the shoulder or on the ankle. The electronic part of the diode dosemeter was fitted on the belt. The connecting cable was fitted on the body or feet to reduce its mechanical stress. Eight comparisons consisting of CA, PTCA interventions and placement of pacemaker were made.

Results and discussion

Tests of diode dosemeter in laboratory conditions

Figures 1 to 3 illustrate the characteristics of the diode dosemeter. For a 4 mmAl filtered beam the response varies within 20% for kV range from 40 to 60 kV (Fig. 1.). The response decreases more deeply towards lower kV values. For an 82 kV x-ray beam an increased response of the diode

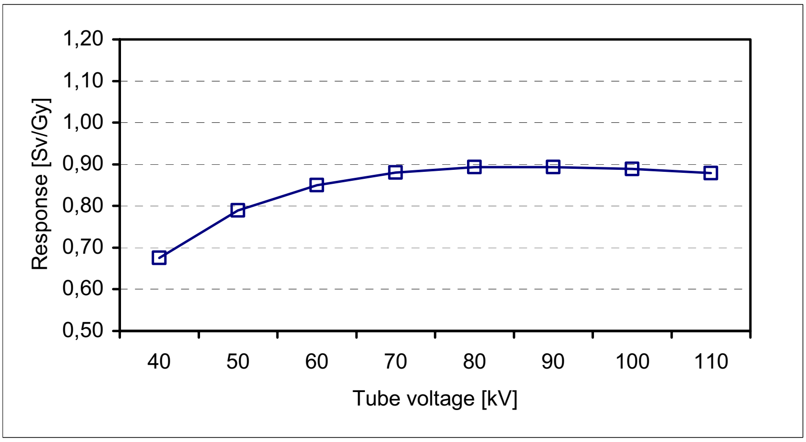

Figure 1. Response of the diode dosemeter for air kerma relative to X-ray tube voltage. For each kV used, the response is the ratio of the displayed $H_p(0,07)$ by the diode dosemeter to the air kerma measured by the reference instrument. Measurements are performed free in air, without a backscatter phantom. Tube filtration 4 mmAl. Air kerma rate 13 – 240 μGy/min. With each kV value the exposed air kerma 213 -240 μGy.

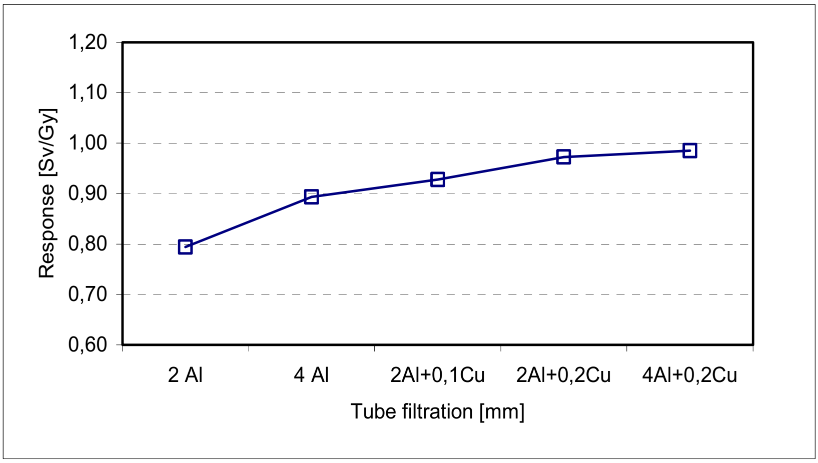

Figure 2. Response of the diode dosemeter for air kerma relative to X-ray quality modified by filtration. The response is the ratio of the Hp(0,07) displayed by the diode dosemeter to the air kerma measured by the reference instrument. Measurements performed free in air, without a backscatter phantom. Tube voltage 82 kV, air kerma rate 60 – 200 μGy/min. With each filtration value the exposed air kerma 60 -230 μGy.

dosemeter is indicated relative to increased beam filtration (Fig. 2.). Totally 20% change in response is indicated relative to beam filtration studied.

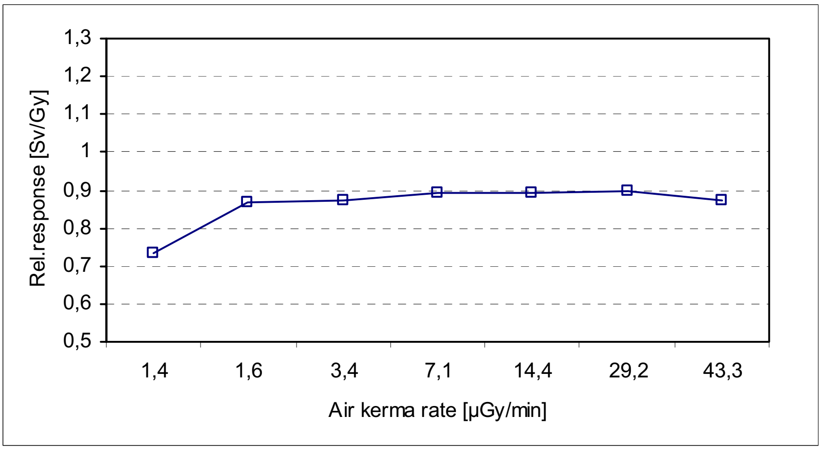

Figure 3. Response of the diode dosemeter for air kerma rate. For each air kerma rate value used, the response is the ratio of the Hp(0,07) displayed by the diode dosemeter to the air kerma rate measured by the reference instrument. X-ray quality: 82 kV, tube filtration 4 mmAl.

Figure 3. shows that the response of the diode dosemeter is independent of the air kerma rate down to 1,6 µGy / min. At lower air kerma rates the response drops steeply down. This may be due to the small detector and its associated low count rate. Further, vigorous shaking of the cable caused noise signals which could disturb remarkably the measuring signal at low values for the air kerma or the air kerma rate. Thus, the meter is not very well suited for very low dose rates and dose measurements behind radiation shields.

Staff doses measured with TL dosemeters

Table 1 illustrates the partial body doses of the operating staff in various procedures. The mean measured TL doses, the DAP values and DAP ranges were about the same for the CA examinations and the PTCA procedures, but were higher for the pacemaker procedures. The extremity doses of the staff varied considerably especially for hands. The shoulder doses were systematically larger for cardiologists than for radiographers. Difficulties to use radiation shields between the patient and cardiologist during some phases of the procedure explain the largest staff doses (up to 2 mGy for the ankle and 2,7 mGy for hand).

Comparison of diode and TL dosemeters for measurement of the staff dose

Figure 4 illustrates the doses measured simultaneously on the operating staff with the diode dosemeter and TL detectors. In six cases the results agreed within 35%. In two cases the values differed by about a factor of 2. The main contribution to the differences may be due to sensitivity and set-up of the

Table 1. The shoulder, ankle and hand doses of the operating staff, measured for 45 procedures using TL-detectors. Mean (max.) and range of DAP values is given for each procedure.

	N		Cardiologist Shoulder mSv	Cardiologist Ankle mSv	Cardiologist Hand mSv	Radiographer Shoulder mSv	Radiographer Ankle mSv	DAP Gycm²
CA	29	mean max.	0,05 (0,17)	0,02 (0,14)	0,05 (0,13)	0,02 (0,06)	0,03 (0,12)	39,4 (6,02-135)(***
PTCA	12	mean max.	0,06 (0,16)	0,02 (0,07)	0,075(* (0,09)	0,02 (0,06)	0,06 (0,12)	38,0 (1,89-126)(***
Pace-maker	4	mean max.	0,12 (0,23)	0,88(* (1,98)	2,7(**	0,09(**	0,30(**	62,3 (13,9-179) (***
All		mean	0,06	0,09	0,27	0,02	0,05	41,1

*) 3 interventions, (** one intervention (***range

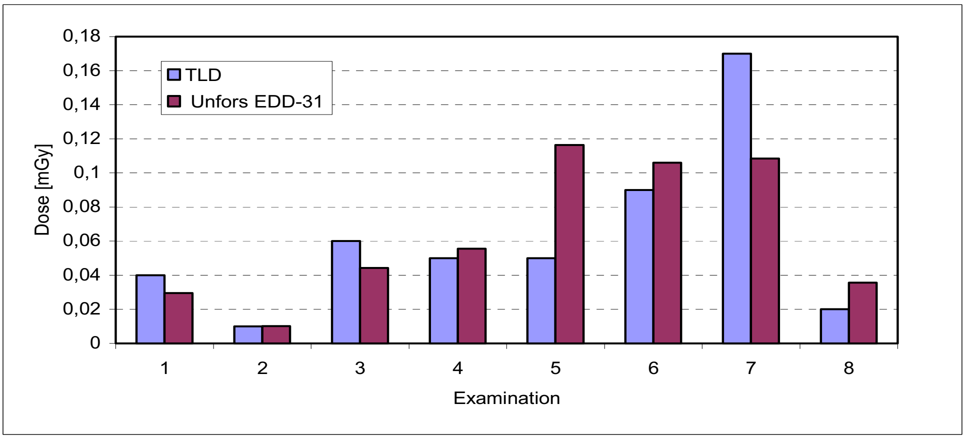

Figure 4. Comparison of doses measured on the cardiologist and radiographer with TL and Unfors EDD-30 detector in 8 different interventional procedures.

detectors at the steep dose gradients behind the personal radiation shields. For diode dosemeter the noise signals caused by the microphonic effects of the cables could cause some uncertainty on the results. Furthermore, some part of the differences can be explained by the separate calibration procedures of the TL dosemeters and the diode dosemeter. A clear advantage of the electronic diode dosemeter is the immediate availability of measurement results after each examination procedure. The electronic diode dosemeters are quite useful for personnel dose monitoring, but they will be more suitable after detector cabel modifications.

Acknowledgements

The study is a part of the DIMOND research project (EC contract n. FIGM-CT-2000-00061).

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7c Radiation dose and dose optimization in paediatric thorax examinations in HUCH Hospital for Children and Adolescents

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Purpose

Nearly 9% of all x-ray examinations in Finland are performed with patients below 16 years of age. The percentage of x-ray examinations at university hospitals for patients below 16 years is about 15 %. Paediatric thorax examinations cover almost 30 % of all children's examinations. The risk of late effects due to radiation is assumed to be 3 – 4 times larger for children than for 30 – 40-year-old adults. The EC has introduced recommendations on quality criteria for paediatric x-ray examinations [1]. The EC Medical exposure directive (MED) 97/43/Euratom [2] requires medical radiation exposure to be determined in regular x-ray examinations and the obtained values to be compared to national reference values. The directive also assumes that the x-ray examination procedure is evaluated and the use of radiation is optimised in order to reduce the exposure. Radiation doses to children should be given special attention[3]. In Finland there are no reference values yet concerning exposure in paediatric x-ray examinations [4, 5]. In a joint quality project of HUCH and STUK the aim is to determine the radiation doses and assess the image quality in paediatric thorax examinations. The purpose is to optimise the x-ray examination procedure thus reducing the radiation exposure to children as much as possible.

Material and methods

The doses of radiation exposure for children of different ages were determined at HUCH Hospital for Children and Adolescents. The breakdown by age was: the new-born, <30 d, 30 d - <1y, 1 - <5y, 5 - <10y, 10 - <16y. Exposure measurements were performed for approximately 300 paediatric thorax examinations. There were about 530 single images in ap, pa and lateral projections. The dose area product, DAP (mGy × cm^2), was measured by a Gammex RMI 840A Dose Area Product Meter and Gammex RMI 841-C chamber. The chamber chosen was an especially sensitive one since the measured dose area product of children X-ray examinations can be much lower than that of the adult patient. The DAP meter was calibrated for various filtrations and tube voltages at the hospital. The entrance surface dose, ESD (mGy), was determined by calculation. For this purpose the air kerma was measured by an ionisation chamber and Radcal 9015 radiation monitor. The field size of the radiation beam was defined from all images. Various imaging techniques (anti-scatter grid, tube voltage, filtration) were tested for optimising the radiation dose and the image quality. The image quality was evaluated from clinical patient x-ray images by radiologists and radiographers using an evaluating indicator specially developed for this purpose. With the aid of PCXMC calculating programme [6] the results were also used to calculate the effective dose to patients of different ages.

Results

The results (DAP and ESD) are represented by age groups. The measured doses were distinctively below the EC recommendation limit. For example, the ESD doses for 5-10-year-olds were: ap 47 µGy and in lat. projection 85 µGy (the EC recommendation for 5-year-olds is 100 µGy and 200 µGy, respectively). The calculated effective dose (ap/pa + lat. projections) caused by this study was 0,02 mSv. By changing the imaging technique (anti-scatter grid, filtering) the radiation dose was decreased by 30 - 60 % from the original while the image quality remained good. The image quality was observed to stay good on average and repetitions were called for very rarely. Without the anti-scatter grid (<5 y) the image quality was good enough, but then again the quality was more affected by the image field size and the imaging values, e.g. the mAs value. The image quality was not satisfactory using 80 k V (1 - <5 y). In examination situation an assistant was needed for all below 5-year-olds and for half of the 5 – 10-year-olds.

Conclusions

Many children need several recurrent x-ray examinations during their treatment that may accumulate to a remarkable radiation dose. University hospitals should pay special attention to the quality of examinations and to

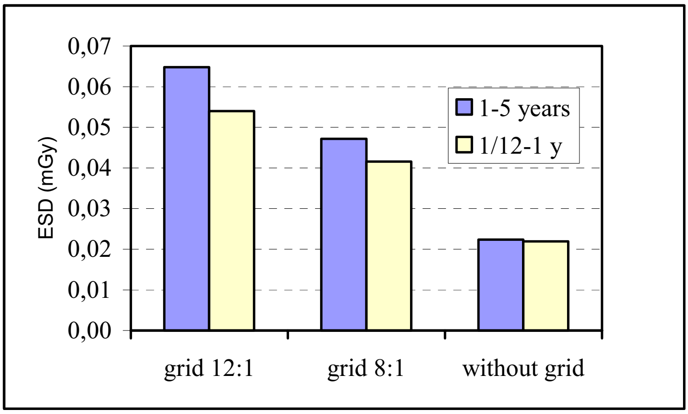

Figure. Entrance surface dose (ESD) in thorax (ap) examinations by changing the imaging technique (with and without grid) for children aged 1/12-<1 and 1 – <5 years.

radiation exposure doses. Using the dose area product measuring the DAP-meter must be sensitive enough for paediatric X-ray examinations. The obtained doses give a great starting point to determine the national reference values for paediatric thorax examinations. The results show that with careful optimising of image quality and other dose-sensitive factors the radiation dose can be considerably reduced in paediatric examinations.

Table. Dose area product (DAP), entrance surface dose (ESD) and calculated effective dose (mSv) in thorax examinations. Comparison with quidelines on best practices [7] and diagnostic reference levels [1].

			Dose-area product (mGy cm^2)			Entrance surface dose (mGy)				Effective dose (mSv)	
Exami-nation (proj.)	**Age group (years)**	**Number of pa-tients**	**Mean DAP**	**Third quartie**	**Cook [7]**	**Mean ESD**	**Third quartile**	**Cook [7]**	**Diagnostic reference level [1]**	**This work**	**Cook [7]**
Chest	< 1/12	15	3,4	4,2	2	0,043	0,045	0,02	0,080	0,01	≤ 0,01
(AP/PA)	1/12 - <1	37	6,2	7,3	3	0,042	0,042	0,02		0,01	≤ 0,01
	1 – <5	40	11	14	5	0,047	0,057	0,03	0,100	0,01	≤ 0,01
	5 – <10	41	20	22	16	0,057	0,060	0,04	0,100	0,01	≤ 0,01
	10 - <16	41	32	39	29	0,060	0,068	0,05		0,01	≤ 0,01
Chest	1/12 - <1	37	9,7	11	10	0,067	0,072	0,06		0,02	≤ 0,01
(Lat)	1 - <5	40	18	22	23	0,085	0,095	0,08	0,200	0,01	≤ 0,01
	5 - <10	41	34	36	23	0,139	0,125	0,08	0,200	0,01	≤ 0,01
	10 - <16	41	87	113	44	0,209	0,255	0,14		0,02	≤ 0,01

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7d Dose-area product and entrance surface dose in paediatric radiography

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Abstract
Dose-area products (DAP) in paediatric radiography were measured in four university hospitals in Finland. The entrance surface dose (ESD) was calculated from the measured DAP value for each radiographic projection. The purpose was to combine the results with other European studies for development of diagnostic reference levels for paediatric X-ray examinations. The study included 740 paediatric patients, and a total of 1500 single projections were recorded, including 660 projections from extremities. Results were compared with recommended best practices and diagnostic reference levels for ESD. Ratios of DAP to ESD were studied to estimate the levels of DAP corresponding to recommended ESD reference levels. It is desirable for practical purposes that diagnostic reference levels for radiographic projections are also expressed in terms of dose-area product.

Introduction
The International Commission on Radiological Protection (ICRP) has recommended *diagnostic reference levels (DRL)*, applying to easily measurable quantities, to be used for the optimisation of protection of medical exposures of patients [1]. The Medical Exposure Directive (MED) 97/43/Euratom [2] introduced the diagnostic reference levels into regulatory use as a tool for optimisation and quality assurance, and required regular assessment of quantities related to patient dose and comparison of these quantities with nationally established diagnostic reference levels. The directive also requested determination of dose distributions and population doses from medical exposures, and focused special attention on the optimisation and quality assurance of paediatric diagnostic examinations.

For monitoring patient dose in radiographic examinations, *the entrance surface dose (ESD)* or *dose-area product (DAP)* can be determined.

For radiographic projections, the diagnostic reference levels are typically available for ESD only. ESD or DAP values for single projections are also needed in calculating organ doses for risk assessment or effective dose determination. Special reference dose levels are needed for paediatric patients, at least for the smallest children, because optimal patient doses depend strongly on patient thickness, and because estimated radiation risks to children are typically higher than to adults receiving equal organ doses. It is, however, rather complicated to establish diagnostic reference levels for paediatric patients because of wide ranges of patient size.

European recommendations for DRL values for paediatric patients have been published in European guidelines on quality criteria for diagnostic radiographic images [3] and in the Radiation Protection series [4]. Development of diagnostic reference levels in paediatric radiology, including computed tomography and fluoroscopic examinations, has been discussed in European research project on radiation protection in paediatric X-ray examinations [5-8]. Results from dose-area product measurements during the project in Finland for radiographic projections are reported in this study. Results from fluoroscopic examinations among paediatric patients in Finland have been published earlier [9].

Material and methods

Dose-area products among paediatric patients in radiographic examinations were measured at Helsinki, Turku, Oulu and Kuopio university hospitals. Patient age, sex, height and weight were recorded for each examination and, for each projection, the measured DAP value and relevant examination techniques, such as filtration, tube voltage, current-time product, field size on the film, focus to film distance and focus to skin distance. Corresponding entrance surface dose (ESD) was calculated from the DAP value, multiplying it by the backscattering factor and dividing by the field area at the entrance surface plane. The dose-area products were measured using a transmission ionisation chamber (RMI Gammex DAP 840) which was calibrated at the X-ray laboratory of STUK and at the hospitals for various filtrations and tube voltages [9].

The study in the four hospitals included a total of 740 paediatric patients, who were less than 16 years old at examination. A total of 1500 single radiographic projections were recorded, including 660 projections from extremities. Patients were classified into four age groups: under 1 year, 1 - 4 years, 5 - 9 years and 10 - 15 years. Because the number of subgroups was high, the number of patients was relatively low in many subgroups. The mean, minimum and maximum values of DAP and ESD were determined, and also the standard deviation, median and third quartile when relevant. Results were compared with British guidelines on best practices [10] and European recommendations for ESD reference levels [3,4]. Ratios of mean DAP to mean ESD were studied to estimate the levels of DAP corresponding to recommended ESD reference levels.

Results

Results from 15 examinations and 27 radiographic projections are collected in Table 1. The number of patients, and the mean values of patient age, height and weight, dose-area product and entrance surface dose are shown for each projection and age group. For comparison, the DAP and ESD values corresponding to the best practice [10] are shown for each subgroup. European DRL recommendations [3,4] for standard 5-year-old patients are shown at age groups of 1 - 4 years and 5 - 9 years, according to the guidance [3] of applying the same levels to both groups. These diagnostic reference levels are only available as ESD values, not as dose-area products.

The ratios of DAP to ESD, calculated from the best practice DAP and ESD values [10] and from the mean values of this study, are shown to allow some comparison and assessment of dose-area products corresponding to established ESD diagnostic reference levels. Among paediatric patients the DAP/ESD ratio typically increases with increasing age because it is proportional to X-ray field area; the ratio is equal to the quotient of the area and the backscattering factor. Among the four age groups the DAP/ESD ratio typically ranges, approximately, from 150 cm^2 to 300 cm^2 for skull, to 400 cm^2 for chest lateral and to 600 cm^2 for chest AP projections. The range is about 200 - 600 cm^2 for abdomen, 100 - 300 cm^2 for lumbar spine, 100 - 200 cm^2 for knee, 50 - 150 cm^2 for ankle and elbow, and 50 - 100 cm^2 for wrist examinations. The pactices and projections used in hip and pelvis examinations in the Finnish hospitals are not directly comparable with the guidelines on best practice [10].

The values of DAP and ESD vary over a large scale and have a skew, typically lognormal distribution. Calculated value of linear standard deviation is thus typically high, and the mean value is typically between median and third quartile. Typical mean ESD values resulting from this study are lower than European DRL recommendations [3,4]. Measured DAP values are also typically lower than corresponding ESD reference levels multiplied by an estimated DAP/ESD ratio related to age, examination and projection.

Table 1. Dose-area product (DAP) and entrance surface dose (ESD) in paediatric radiographic examinations. Comparison with best practices [10] and diagnostic reference levels (DRL) [3,4].

Examination	Patients					DAP (mGy cm²)		ESD (mGy)			DAP/ESD (cm²)	
(projection)	Age group (years)	Number of patients	Mean age (years)	Mean height (cm)	Mean weight (kg)	Mean, this work	Best practice [10]	Mean, this work	Best practice [10]	DRL [3,4]	This work	Best practice [10]
Skull	< 1	8	0,28	59	5,9	95	22	0,61	0,15		156	147
(AP)	1 - 4	5	3,4	99	16	221	80	0,96	0,48	1,5	230	167
	5 - 9	5	8,2	120	24	342	110	1,07	0,73	1,5	320	151
	10 - 15	3	11,1	138	40	352	204	1,27	0,94		277	217
Skull	< 1	7	0,26	58	5,6	81	14	0,45	0,09		180	156
(Lat)	1 - 4	5	3,4	99	16	145	53	0,56	0,30	1,0	258	177
	5 - 9	3	7,3	112	20	196	60	0,73	0,36	1,0	268	167
	10 - 15	3	11,1	138	40	273	112	0,81	0,46		337	243
Chest	newborn											
(AP)	0,5-2 kg	29	0,05	37	0,9	3,4	2	0,04	0,02	0,08	91	100
Chest	< 1	37	0,44	62	6,4	8,9	3	0,05	0,02	0,10	181	150
(AP/PA)	1 - 4	61	2,7	91	13,5	22	5	0,07	0,03	0,10	299	167
	5 - 9	24	7,1	123	24	33	16	0,08	0,04	0,10	417	400
	10 - 15	31	13	155	47	71	29	0,12	0,05		586	580
Chest	< 1	33	0,47	64	6,6	12	10	0,07	0,06		171	167
(Lat)	1 - 4	60	2,7	91	13,5	35	23	0,14	0,08	0,20	250	288
	5 - 9	23	7,1	123	24	59	23	0,17	0,08	0,20	343	288
	10 - 15	30	12,9	155	47	124	44	0,27	0,14		459	314
Chest,	< 1	55	0,25	56	5,0	3,4	3	0,021	0,02	0,10	164	150
heart	1 - 4	8	2,3	88	13,3	26	5	0,08	0,03	0,10	311	167
(AP/PA)	5 - 9	7	6,6	115	20	34	16	0,09	0,04	0,10	368	400
	10 - 15	9	13	153	48	96	29	0,149	0,05		647	580
Chest,	< 1	54	0,26	56	5,1	5,3	10	0,033	0,06		160	167
heart	1 - 4	8	2,3	88	13,3	42	23	0,16	0,08	0,20	259	288
(Lat)	5 - 9	7	6,6	115	20	48	23	0,16	0,08	0,20	298	288
	10 - 15	9	13	153	48	149	44	0,33	0,14		445	314
Abdomen	< 1	4	0,39	62	5,4	27	9	0,11	0,05		245	180
(AP)	1 - 4	8	3,3	96	16	145	30	0,40	0,16	1,0	363	188
	5 - 9	1	7,9	125	25	248	74	0,39	0,25	1,0	631	296
	10 - 15	6	12,4	151	49	1275	360	2,06	0,66		619	545
Urography	< 1	7	0,42	61	6,1	181		1,17			155	
(AP)	1 - 4	10	2,7	93	14,4	456		2,1			218	
	5 - 9	5	7,0	123	28	819		2,85			287	
	10 - 15	7	12,8	149	42	1846		4,0			465	
Lumbar	< 1	1	0,55	59	5,1	27	10	0,29	0,19		92	53
spine	1 - 4	4	2,2	88	12	105	48	0,49	0,37		214	130
(AP)	5 - 9	5	7,5	133	33	234	232	1,08	0,98		217	237
	10 - 15	7	13,3	154	48	764	541	2,43	1,75		314	309
Lumbar	< 1	1	0,55	59	5,1	69	12	0,73	0,14		95	86
spine	1 - 4	4	2,2	88	12	120	104	0,65	0,70		185	149
(Lat)	5 - 9	5	7,5	133	33	552	300	3,19	1,52		173	197
	10 - 15	8	13,3	151	46	1346	2216	5,22	8,46		258	262
Hip	< 1	12	0,53	65	7,5	12	5	0,10	0,07		120	71
(AP)	1 - 4	14	2,6	89	12,6	47	68	0,27	0,22		174	309
	5 - 9	15	6,8	118	23	91	150	0,53	0,42		172	357
	10 - 15	23	12,6	152	48	314	292	1,24	1,13		253	258
Hip	1 - 4	8	3,7	102	18	66	68	0,47	0,22		140	309
(Med.lat)	5 - 9	9	7,0	123	26	61	150	0,42	0,42		145	357
	10 - 15	20	12,5	152	48	243	292	1,11	1,13		219	258
Pelvis	< 1	5	0,5	68	8,3	13	5	0,09	0,07	0,20	144	71
(AP)	1 - 4	10	2,4	90	14	53	68	0,26	0,22	0,90	204	309
	5 - 9	5	7,4	121	23	170	150	0,57	0,42	0,90	298	357
	10 - 15	4	13,3	154	55	1743	292	2,80	1,13		623	258

Table 1. Continues.

Examination	Patients					DAP (mGy cm²)		ESD (mGy)			DAP/ESD (cm²)	
(projection)	Age group (years)	Number of patients	Mean age (years)	Mean height (cm)	Mean weight (kg)	Mean, this work	Best practice [10]	Mean, this work	Best practice [10]	DRL [3,4]	This work	Best practice [10]
Femur	1 - 4	5	2,7	93	21	17		0,08			205	
(AP)	5 - 9	4	7,4	122	25	85		0,24			356	
	10 - 15	10	14,2	156	57	246		0,73			338	
Femur	1 - 4	3	3,2	102	29	19		0,09			222	
(Lat)	5 - 9	4	7,4	122	25	66		0,19			345	
	10 - 15	6	14,1	166	56	116		0,31			380	
Knee	< 1	1	0,93	71	8,5	2,5	8	0,03	0,09		83	89
(AP)	1 - 4	1	4,5	106	18	11	11	0,09	0,09		121	122
	5 - 9	4	7,9	129	29	19	20	0,17	0,12		114	167
	10 - 15	22	14,6	163	56	110	19	0,51	0,10		214	190
Knee	< 1	1	0,93	71	8,5	1,3	8	0,015	0,07		83	114
(Lat)	1 - 4	1	4,5	106	18	10	12	0,08	0,08		122	150
	5 - 9	3	8,0	132	29	13	20	0,08	0,12		164	167
	10 - 15	29	14,5	163	56	122	20	0,53	0,10		230	200
Lower leg	< 1	3	0,51	69	8,7	2,5		0,02			124	
(AP)	1 - 4	8	2,9	92	14	12		0,06			195	
	5 - 9	10	7,7	126	26	17		0,08			228	
	10 - 15	27	13,9	161	51	29		0,08			345	
Lower leg	< 1	2	0,51	69	8,7	5,6		0,034			166	
(Lat)	1 - 4	8	2,8	91	14	12		0,06			205	
	5 - 9	12	8,0	127	27	18		0,08			232	
	10 - 15	25	13,9	161	51	33		0,09			352	
Ankle	1 - 4	4	3,4	96	14	2,4	5	0,028	0,08		86	63
(AP)	5 - 9	5	7,6	127	26	18	10	0,14	0,09		129	111
	10 - 15	11	14,0	156	46	35	15	0,22	0,11		159	136
Ankle	1 - 4	4	3,5	92	12,5	4,9	5	0,05	0,08		105	63
(Lat)	5 - 9	4	7,6	129	26	15	10	0,20	0,09		76	111
	10 - 15	12	13,9	155	45	60	15	0,29	0,11		209	136
Elbow	1 - 4	5	3,5	103	17	11	4	0,10	0,08		105	50
(AP)	5 - 9	10	7,5	125	25	6,4	8	0,05	0,09		124	89
	10 - 15	7	13	152	43	12	10	0,12	0,11		102	91
Elbow	1 - 4	6	3,5	103	17	10	4	0,10	0,08		96	50
(Lat)	5 - 9	10	7,5	125	25	5,7	12	0,05	0,09		110	133
	10 - 15	7	13	152	43	12	15	0,12	0,10		97	150
Forearm	1 - 4	6	2,4	80	10	12		0,10			123	
(AP)	5 - 9	14	8,3	128	27	13		0,08			165	
	10 - 15	8	13	142	35	12		0,06			198	
Forearm	1 - 4	5	2,3	78	9	3,8		0,03			109	
(Lat)	5 - 9	13	8,4	128	28	12		0,07			165	
	10 - 15	8	13	142	35	11		0,06			184	
Wrist	< 1	1	0,8	70	8,3	1,3	2	0,02	0,06		60	33
(AP)	1 - 4	1	4,9	113	19	6,4	4	0,06	0,07		103	57
	5 - 9	16	6,9	125	26	5,0	6	0,05	0,08		97	75
	10 - 15	26	13	155	46	6,3	8	0,06	0,09		103	89
Wrist	1 - 4	1	4,9	113	19	5,6	4	0,07	0,07		80	57
(Lat)	5 - 9	16	7,0	125	26	4,8	6	0,06	0,09		79	67
	10 - 15	23	13	156	47	6,1	8	0,07	0,09		94	89

[a] Finnish diagnostic reference level for adult patients, when different from European DRL [3,4]

In most examinations the mean DAP and ESD values of this study are higher, by a factor of 2 to 3, than the values from the guidelines on best practice [10]. In some subgroups, such as skull lateral (under 1 year) and knee projections (10 - 15 years), the factor is as high as 5. There are also subgroups in which the Finnish mean values are slightly lower than the values from ref. [10]. In most of these cases the measured DAP value is very low and may thus have been affected by large uncertainties due to a nonlinear response of the DAP meter and insufficient resolution of reading.

Among the subgroups of Table 1, there are a total of 77 subgroups whose mean values result from two or more hospitals. The mean DAP or ESD of a single hospital was more than twice the total mean in 21 subgroups: 4 in elbow, hip and chest projections, 3 in ankle and pelvis, and 1 in forearm, lumbar spine and chest-heart projections. The distribution of these cases among the four participating hospitals was 1, 2, 5 and 13 cases per hospital. In 14 cases the high value was the only result from that hospital in the subgroup.

Conclusions

Comparisons of the measured dose-area products and derived entrance surface doses between hospitals and with recommended diagnostic reference levels [3,4] and best practices [10] seem to demonstrate that patient doses in paediatric radiography can be reduced if optimised imaging techniques are used. It is thus important to establish national reference levels or other guidance for the X-ray examinations of paediatric patients. Dose dependence on patient thickness and wide ranges in patient size are major problems in developing reference levels for paediatric radiology [7,8]. Dose-area product depends on patient size more strongly than ESD.

As the dose-area product is proportional to the radiation energy imparted to the body, it may represent the exposure of patient to radiation even better than the entrance surface dose. For dose-area product measurements among the smallest children the sensitivity, linearity and calibration of the DAP meter are important. Entrance surface dose can be derived from the measured dose-area product if the field area at the entrance surface plane is determined accurately, which may be difficult in practice. To avoid repeated extra calculations from DAP to ESD, the diagnostic reference levels should also be established in terms of dose-area product.

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7e The effect of image receptor change on radiation exposure to patients in the intensive care of chest X-ray examinations

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Introduction

Digital imaging is becoming increasingly popular in radiology. The use of imaging plates is now common even in small X-ray departments and health centers. The effect of the imaging plates on dose and image quality has been studied and compared with the conventional film-screen system[1-5]. In the beginning, the change in image receptor may cause many problems with regard to the radiation dose to patients. With a film-screen system, the optical density of the film effectively limits the dose to the patient, but in digital imaging this limitation does not exist. When a film-screen system is changed to a digital imaging plate system, there is no evidence indicating whether the dose is increasing or decreasing. In this study, the dose to the patient in bedside chest examinations was studied over a five-year period.

Materials and methods

The measurements were made in a university hospital over a five-year period. The image receptors were changed during the study period as follows: a) conventional film-screen system (Kodak Lanex Medium and T-Mat E); b) Fuji's imaging plates with small film size (digital 1); c) Fuji's imaging plates with large film size (digital la and digital 1b); and d) Agfa's IP imaging plate (digital 2). The staff using the X-ray unit was mainly the same throughout this period[6]. In all, 507 chest ap X-ray examinations were performed for patients (aged 18–90 years, mean age 58.4 years) in the intensive care unit by means of the mobile X-ray unit. The surface dose (ESD) was calculated on the basis of the radiation output of the X-ray tube and the examination techniques[7]. Effective doses were calculated by using the ODS-60 program[8].

Table 1.

Image receptor	BMI Average	ESD (mGy)		E_{eff} (mSv)	
		Average	Range	Average	Range
film-screen	29,9	0,67	0.26-1.03	0.14	0.054-0.25
digital1	26	0,74	0.38-1.37	0.16	0.085-0.27
digital1a	17,1	0,82	0.22-1.52	0.17	0.069-0.36
digital1b	26,5	1,07	0.39-3.39	0.20	0.07- 0.67
digital2	24,7	0,66	0.27-1.30	0.12	0.03- 0.26

BMI=body mass index (kg/m^2)

Results

Table 1 shows the entrance surface doses (ESD) and effective doses (E_{eff}) with various image receptors.

The radiation dose to the patient observed in this study is rather high compared against the European guidelines (0.3 mGy in chest pa). The main reason for this is the short focus-film distance (range 88–130 cm, average 103.6 cm) when the mobile X-ray unit is used. The total filtration was 3 mmAl. The average radiation dose to the patient increased after changing to a new digital image receptor except in the case of the most recent change. The average body mass index (BMI) with various image receptors are shown in table 1 and in Figure 2. The exposure area in examinations of female patients became wider when using all the digital systems except the last receptor (width of the field: conventional 35.8 cm; digital I 38.4 cm; digital Ia 40.4cm; digital Ib 39.5 cm; and digital 2 33.7 cm). The size of the exposure area on the image receptor was measured by the radiographer who made the chest - x-ray examination.

According to the recommendations of STUK[9], for being included in dose measurements for establishing DRL or chyecking compliance with them the patient's weight should be between 55 and 85 kg, average 70 kg. The data for this study comprised 335 patients weighing from 55 to 85 kg. The trend of the dose was still the same as for all patients (Figure 1).

Conclusion

Introduction of a new image receptor requires a lot of work, especially in the case of digital imaging plates, which are sensitive to imaging techniques. The results show that average radiation doses to patients increased rather than decreased after a new digital imaging receptor was taken into use. One reason for this may be the high latitude of the imaging plates. The limitation of radiation field size, the use of a grid and the noise level in image quality are also affecting the dose level.

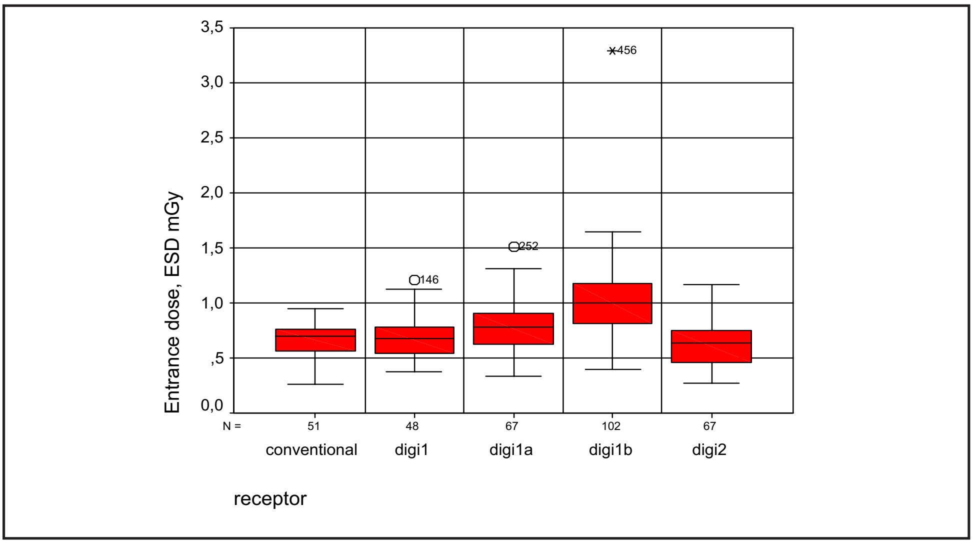

Figure 1. ESD for different receptors with patients of "normal" size.

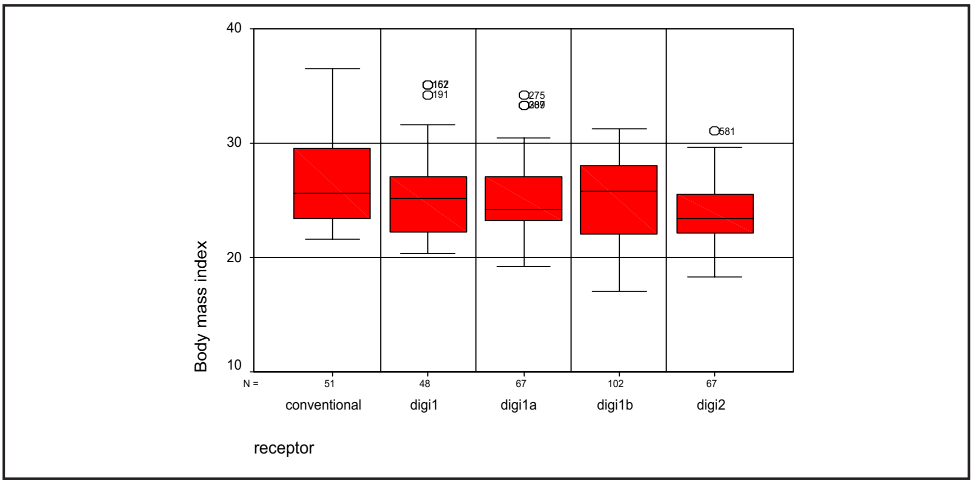

Figure 2. Body mass index of the "normal" sized patients with different image receptors. As seen in Figure 2, the patients' BMI was nearly the same throughout the study, and was even the lowest for the last digital image receptor.

The use of digital imaging also places new demands on radiographers. The portable chest radiograph remains one of the simplest examinations used at the bedside to evaluate the cardiopulmonary status of patients and the position of lines, tubes and monitoring devices[10]. According to a study by Wagner et al[11], 60% of patients in intensive care units receive life support therapy, 35% have central venous lines, 18% have pulmonary artery catheters and 38%

receive mechanical ventilation. These facts make x- ray examinations of the seriously ill patients in intensive care units a very demanding task. Lack of education and training in the use of digital imaging plates at the beginning of the 1990s caused many problems with imaging techniques. The image quality was not as good as the previous image quality. The radiation dose was therefore increased in order to improve the image quality. In addition, the radiation field was not limited as strictly as before; this also meant increasing radiation doses. As the years passed, radiographers got more information on and experience of how to use the imaging plate system, enabling them to find the optimal imaging technique for the bedside chest examinations. Now, after the five-year period, the dose to the patient is at the same level with imaging plates as it was earlier with a conventional film-screen system.

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7f Need for radiological education among the personnel performing X-ray examinations in health centres

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Abstract

Health centres play an important role in performing X-ray examinations in Finland because about 40 % of X-ray examinations are done there. In some small health centres, people who lack the proper radiation protection education are performing X-ray examinations. The work of health care personnel conducting X-ray examinations and their need for education and training in radiation protection were studied in 16 health centres. Other than radiographers, about 60% of the staff members participating in this study replied that their education was not adequate and that they would need additional education and training.

Introduction

The Medical Directive (97/43/Euratom)[1] sets special requirements for health care personnel's education and training in radiation protection. Recommendations concerning continuing education and training after qualification and after the introduction of new techniques have been put forward by the European Commission[2]. In Finland, some 250 health centres conduct about 40% of all X-ray examinations and roughly 42% of all paediatric X-ray examinations[3]. In the health centres, the X-ray studies are conducted by the radiographers, registered nurses, practical nurses, ambulance attendants and some other staff groups. Some small health centres delegate X-ray-related tasks to staff members who have no formal radiation protection education aside from being experienced in their work.

Health centres are responsible for reporting on various aspects of the organisation of radiation safety – including staff competence, activities, the qualifications of the operating personnel and the distribution of responsibilities – to the radiation protection authority. The purpose of this study was to gather information on the use of staff members lacking the proper radiation safety education to perform X-ray examinations in health centres.

Material and methods

The tasks of health care personnel conducting X-ray examinations and their need for education and training in radiation protection were studied in 16 health centres[4]. The adequacy, experience and level of staff members' formal knowledge of radiation protection, their experience and preparedness for X-raying, quality control and patient protection, and staff members' need for continuing radiation protection education and their willingness to participate in such training were studied among health centre personnel by means of questionnaires. The quality assurance, radiation dose to patients or image quality performed in practice were not checked in this study.

According to the survey carried out in the health centres concerned, X-ray examinations are conducted by 16 radiographers, 6 registered nurses, 8 practical nurses, 12 ambulance attendants and 3 health centre assistants; altogether 45 people.

Results

The personnel participating in this study comprised a total of 33 staff members, who were distributed as follows: radiographers, 46%; registered nurses, 18%; hospital and ambulance attendants, 21%; and practical nurses, 9%. Radiographers conducted about 64% of all X-ray examinations, the others 36%.

Altogether about 66,900 examinations a year are carried out in these health centres. On average, 4460 studies were conducted per health centre. The annual number of examinations carried out in a health centre ranged from 1000 to 7500.

Figure 1 shows the total number of examinations and the number of examinations performed by radiographers in the health centres included in this study. In 10 of the health centres, radiographers carried out almost all of the examinations while in the other health centres, a considerable proportion of examinations was carried out by other staff.

According to the results based on the respondents' answers, staff radiographers reported that their education and training was adequate. Among the respondents who were not radiographers, 61 % reported that their education was not adequate, 33% was satisfied with their radiation protection education, and about 6% didn't respond to this question.

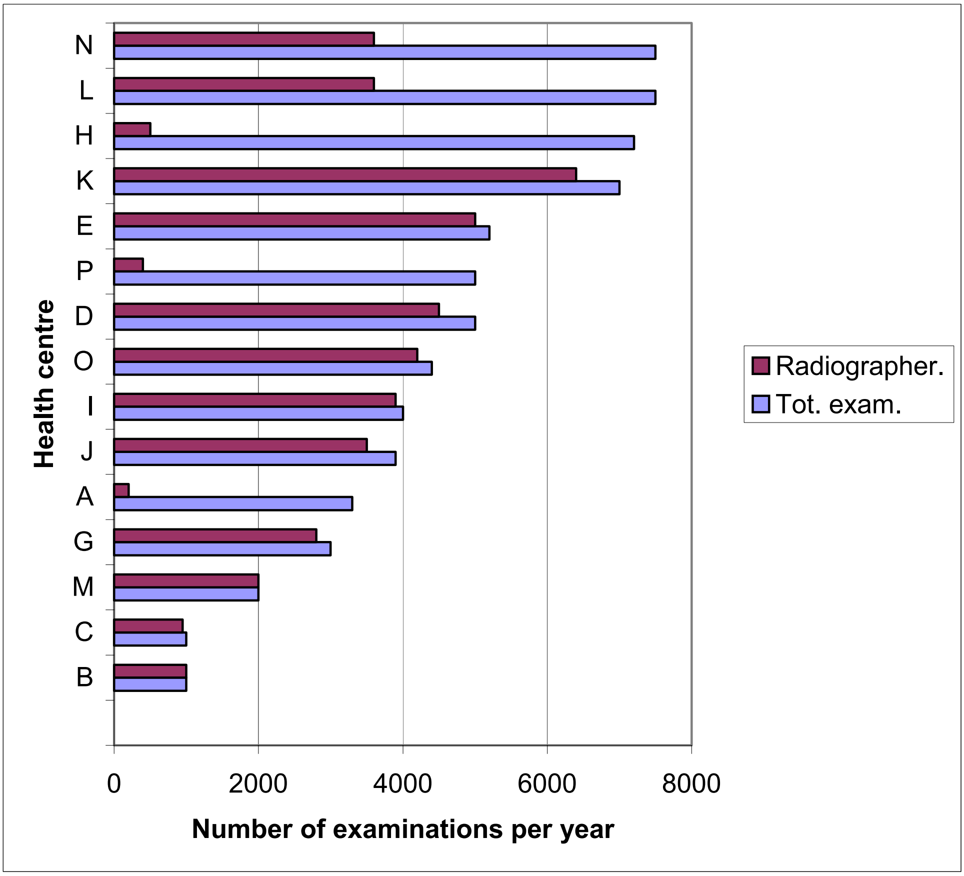

Figure 1. Total number of X-ray examinations per year and number of examinations carried out by radiographers in various health centres.

Figure 2 shows the combined average values for the responses concerning the level of knowledge with regard to X-ray units, radiation protection and imaging techniques for the various personnel groups.

The figure 2 shows clear difference in the average values for the levels of knowledge among the various personnel groups. According to the personnel's own assessment, radiographers had the best level of knowledge and practical nurses had the lowest level of knowledge.

The As seen in figure 1 in some health centers registered nurses, hospital and ambulance attendants or practical nurses made 50% of the x-ray examinations. In other places they made only a small amount during evenings, nights, weekends and radiographer's holiday. They made chest, bone and sinus x-ray examinations. In addition the ambulance attendants made also lateral skull and OPTG -examinations. In all groups of no-radiographer the number of X-ray

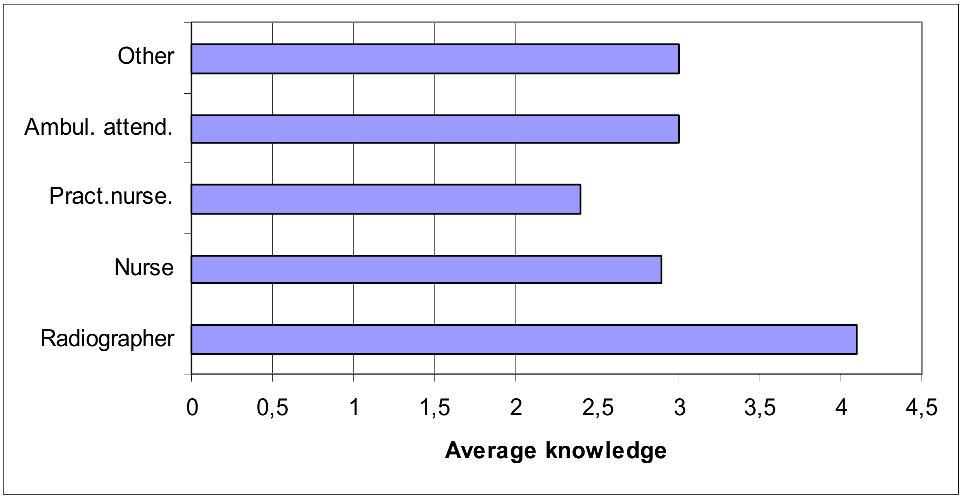

Figure 2. Combined average values for the responses concerning the level of knowledge with regard to X-ray units, radiation protection and imaging techniques for the various personnel groups.

examinations is approximately on the same level, but the amount of the persons in each varies. That's why some registered nurses and hospital and ambulance attendants do not make very often X-ray examinations.

Conclusions

Health centres play an important role in performing X-ray examinations in Finland because about 40% of X-ray examinations are done there. In some small health centres, people who lack the proper radiation protection education are performing X-ray examinations. Excluding the radiographers, about 60% of the personnel groups included in this study_reported that they lacked adequate education and they need additional education and training. The personnel with the lowest qualifications did not know much about the theory and basic concepts of imaging.

The current situation poses many questions. The health centre's medical doctor (who is not a radiologist) is the radiation safety officer responsible for the safe use of radiation at the health centre. This appears to cause some problems in providing the personnel with radiation safety knowledge. One topical task would be to provide continuing education and training in radiation protection for health centre personnel. The ultimate goal is to provide all health centres with radiographers for conducting X-ray examinations. Although the topic of this study is not a general problem, it is important that the issue is recognised and that all possible efforts are made to resolve it.

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7g Radiation doses to the unborn child at diagnostic examinations in Sweden

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Introduction

The use of ionising radiation in a medical examination of a woman caring a child is not always possible to avoid. The following situations can occur:

1. The pregnancy of the patient is known and the examination has to be performed due to medical reason
2. The pregnancy of the patient is unknown at the time of examination

Methods to identify pregnant women at radiological departments in Sweden are already in use, but national rules and methods to calculate the individual dose to the unborn child for different examinations are less evaluated. There is a need of standards for the calculations, estimations and documentation of the radiation dose to the unborn child. According to directives from the European Commission (EUR 16260 En, 1995 and EUR 16262 EN, 1998), every X-ray examination has to be justified and optimised.

The aim of this study is to determine the absorbed dose to the unborn child for common radiation diagnostic examinations used in Sweden and to find a standardised method for dose calculations.

Material and methods

Conventional X-ray

The absorbed dose to fetus has been calculated and measured for the following examinations: chest, bedside chest, lumbar spine, abdomen, hip, pelvis, urography, and barium enema at Linköping University Hospital and County Hospital Ryhov. The absorbed dose to the uterus has been used for simulation of the dose to the fetus. This approximation can be used for a fetus smaller than 12 weeks (3 months). The fetus size is starting to change more rapidly in the weeks 16-20 (after 4-5 months) (ICRP/22/136/01 Draft Report, 2001).

Two available calculation programs have been used WinODS (Rados ODS-60 (WinODS) User guide, 1996 and Servomaa et al, 1989) and PCXMC (Tapiovaara, 1997 and Servomaa and Tapiovaara, 1998). WinODS is based on depth and profile dose curves and PCXMC on Monte Carlo calculated organ doses.

To verify the calculated doses a female human phantom (CIRS ATOM female phantom, Model # 702-004, 2002) was used to simulate the examinations of the patient. The phantom has been loaded with TL-dosimeters ($Li_2B_4O_7$) in order to measure the radiation dose in uterus. The dosimeters have been calibrated in known radiation fields at different energies. The following X-ray units have been used: Siemens Pendex, Philips Pendla Modul and Siemens Siregraph.

Computed tomography

The following examinations are included in this study: chest, abdomen and trauma, both sequence and multi-slice spiral examinations at County Hospital Ryhov. The same phantom was used. Methods described by Nagel (Nagel et al, COCIR, 2000) have been used together with a computer program (WinDose, Kalander et al, 1999) for comparative dose calculations. CT (Computed tomography) units used in this study are Siemens Somatom Plus S and Somatom Plus 4 VZ.

Nuclear medicine

To calculate the fetal dose at nuclear medicine examinations data given in the Fetal dose calculation workbook, RIDIC (Stabin, 1998), ICRP 53 (ICRP 53, 1988) and ICRP 80 (ICRP 80, 1998) have been used. Data on administered activities were obtained from the Linköping University Hospital. Administered activity can show great variation between different hospitals (Mattsson et al, 1998). Reference activities given in European Commission publications (European Commission, Radiation Protection 109, 1999) are not generally used in Sweden. The type of radiopharmaceutical and the possibility of placenta transfer also influence the fetal dose (ICRP 84, 2000).

Results

Conventional X-ray

Table 1 shows the results of calculated and measured fetal doses. Since the DAP (Dose Area Product) value is used as a reference dose value of an examination (European Commission, Radiation Protection 109, 1999) the fetal doses are in this report given in relation to this value as a conversion factor $C_{conv.} = D_{uterus}/DAP$ (mGy/Gycm^2) having the following values:

1. $C_{conv.}$ is < 0.01 if the radiation fields does not include the uterus
2. C_{conv} is approximately equal to 0.5 when the uterus is only partly included

Table 1. Calculations and measurements of the absorbed dose to the fetus and conversion factors $C_{conv.} = D_{uterus}/DAP$ (mGy/(Gycm^2)) for conventional X-ray examinations at Linköping University Hospital and County Hospital Ryhov. For comparison values from ICRP 84 (ICRP 84, 2000) are shown.

Examination	DAP value Gycm^2	PCXMC calculated dose to uterus mGy	WinODS calculated dose to uterus mGy	Measured dose to uterus mGy	Fetal dose ICRP 84 Mean/Max (United Kingdom) mGy	Conversion factor $C_{conv.}$ mGy/(Gycm^2)
Abdomen (Linköping UH)	0.79	0.40	0.38	0.44	1.4/4.2	0.56
Abdomen (Ryhov)	1.28	0.84	0.92	0.63	1.4/4.2	0.49
Chest bedside (Linköping UH)	0.028	<0.001	<0.001	0	< 0.01	0
Chest (Ryhov)	0.086	<0.001	<0.001	0.001	< 0.01	0.01
Hip frontal (Linköping UH)	0.43	0.02	0.3	0.12		0.28
Lumber spine (Linköping UH)	2.71	1.49	2.08/1.38*	1.38	1.7/10	0.51
Lumber spine (Ryhov)	4.15	2.01	1.51	1.75	1.7/10	0.42
Pelvis (Linköping UH)	1.22	0.67	0.9	0.72	1.1/4	0.59
Pelvis (Ryhov)	1.28	0.85	1.06	0.66	1.1/4	0.52
Urography (Ryhov) (10 images)	6.01	4.0	4.6	2.8	1.7/10	0.47
Barium enema (Ryhov) (fluoro + 13 images)	20.2	12.3**	13.0**	7.8	1.1/5.8	0.39

*A small change in the calculation of the position of the radiation field gives large change in fetal dose value.
** Uncertain simulation.

3. C_{conv} is approximately equal to 1.3 if all radiation fields include uterus.

The uncertainty in the evaluation of the conversion factor is less than ± 30 %.

For examinations where the fetus is not included in the primary examination fields a simplified general estimation can be done. The dose is then expected to be less than 1-2 mSv.

In comparison with values of fetal doses for some common examinations in United Kingdom (ICRP 84, 2000) this study show lower values for abdomen and pelvis examinations, probably depending on different sensitivity of film screen system. The higher value for barium meal in this study must depend on different examination procedures. Otherwise the dose levels were similar. A comparison between the used computer programs and the measured values shows that the PCXMC slightly underestimate the dose when the uterus is outside the radiation field, while WinODS slightly overestimate it. When the uterus is included in the radiation field both programs overestimate the dose.

Computed tomography

Table 2 shows the results. The relation between $CTDI_w$ value (weighted Computed Tomography Dose Index) and fetal dose is can be used as a conversion factor $C_{CT} = D_{uterus}/CTDI_w$ (mGy/mGy). The following values are estimated:

1. C_{CT} is < 0.01 if the examined volume does not include the uterus
2. C_{CT} is approximately 1.0 for examinations where the examined volume includes the uterus

The uncertainty in the evaluation of the conversion factor is less than ± 20 %. Agreements between calculations and measurements are also shown. The dose levels given by ICRP 84 (ICRP 84, 2000) (Table 4) for common CT examinations in United Kingdom are in agreement with measured values in this study.

Nuclear medicine

The results of dose calculations for the most common examinations of pregnant patients at the Linköping University Hospital are shown in Table 3. For one examination placenta transfer is included. Comparison with values given in ICRP(ICRP 84, 2000) shows that administered activities in this study are generally lower, which also gives lower fetal dose.

Discussions

The knowledge of doses from different radiation fields and radiopharmaceuticals gives the opportunity to optimise the examination of a pregnant patient in order to minimum the risk to the unborn child. For dose estimates later in the pregnancy (after 3 months) other organs should be included, for example the GI-tract.

Table 2. Calculations and measurements of the absorbed dose to fetus and the conversion factor $C_{CT} = D_{uterus}/CTDI_w$ (mGy/mGy) for computed tomography examinations at county hospital Ryhov. Tabulated $CTDI_w$ values are used in the calculations with Somatom Plus S and measured with Somatom Plus 4 VZ. For comparison values from ICRP 84 (ICRP 84, 2000) are shown.

Examination	$CTDI_w$ mGy	WinDose calculated dose to uterus mGy	COCIR calculated dose to uterus mGy	Measured dose to uterus mGy	Fetal dose ICRP 84 mean/max (United Kingdom) mGy	Conversion factor C_{CT} mGy/mGy
CT abdomen sequence 2: lower abdomen. Somatom Plus S	12	13.7	14.1	13.8*	8.0/49	1.15
CT chest. Somatom Plus S	12	0.022	0.5	0.21	0.06/0.96	0.02
CT trauma sequence 4: abdomen Somatom Plus 4 VZ	14.9	18.2	14.9	15.8*	8.0/49	1.06
CT abdomen. Somatom Plus 4 VZ	15.5	24.8	15.6	15.6	8.0/49	1.01

*The value is not corrected for contributions from other sequences in the same examination not included uterus.

Table 3. Common nuclear medicine examinations used for pregnant women at Linköping University Hospital. Values in brackets represent administered activity used if the pregnancy in known. For comparison values from ICRP 84 (ICRP 84, 2000) are shown. Shaded area includes placenta transfer in the calculation.

Examination	Nuclid	Administered activity ICRP 84 MBq	Administered activity this study MBq	ICRP 84 early preg. dose mGy	This study early preg. dose mGy	ICRP 84 9 months preg. dose mGy	This study 9 months preg. dose mGy
Bone scan	^{99m}Tc	750	600	4.6-4.7	3.12	1.8	1.5
Leucocyte scan (HMPAO)	^{99m}Tc		200		1.74		0.72
Lung perfusion (MAA)	^{99m}Tc	200	75 (50)	0.4-0.6	0.21 (0.14)	0.8	0.30 (0.20)
Lung ventilation (Technegas)	^{99m}Tc		120		Uterus: 0.04		
Kidney (DTPA)	^{99m}Tc	750	200	5.9-9.0	2.4	3.5	0.94
Kidney (MAG3)	^{99m}Tc		75		1.35		0.39

More measurements and simulations are therefore needed in order to determine more exact conversion factors.

The DAP dose meter need to be regularly calibrated and the CTDI values regularly checked in order to obtain reliability estimations. The fetal dose can be manipulated in X-ray and CT examinations by changing views and parameter settings in order to avoid direct exposure of the uterus. For nuclear medicine examinations careful choice of radiopharmaceutical and use of lower administrate activity can substantially lower the absorbed dose to the fetus. The examination time then normally has to be increased to obtain sufficient image quality.

Acknowledgements

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7h Prenatal Radiation exposures at diagnostic procedures: methods to identify exposed pregnant patients.

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Introduction

Knowledge about frequency and doses to embryo/foetus from diagnostic radiology is of great importance both in the sense of estimating the radiation risks but also for optimizing the diagnostic procedures and making decisions regarding alternative procedures. In addition, the pregnant patient has a right to know the magnitude and type of radiation risks expected as a result of foetus exposure. From a risk perspective epidemiological data has shown that the embryo/foetus together with children experience higher radiation sensitivity in terms of induced leukemia and cancer compared to an adult population. Recent estimates give cancer excess lifetime mortality risks for whole body exposures of children and foetus (0-15 y age) of 0.06% (ICRP84, 2000) up to 0.14% per 10 mSv (BEIR-V 1990). In addition to the risk of cancer induction effects of cell killing, e.g. CNS abnormalities, cataracts, malformations, growth retardation, may occur. However, these effects are believed to have a threshold, about 100-200 mGy (ICRP84, 2000), and such foetus doses are rarely reached in diagnostic radiology procedures.

There are 2 principal situations where foetus exposures may occur in diagnostic radiology;

- The pregnancy of the patient is known at the time of examination, but due to the medical indications the examination can not be postponed or put forward in time, and there are no suitable alternative non-radiological procedures.
- The pregnancy of the patient is not known at the time of examination, either due to the fact that the patient is unaware of her pregnancy or the

medical personnel failed to obtain this information. The former situation may occur during the first few weeks from conception, whereas the latter situation may cover a greater gestation period.

The frequency of foetus exposure is not well documented. In Sweden, there are well-established routines to track down pregnant patients before examinations are being performed. However, there are no general obligations or routines to document the cases either (i) when planned examinations are performed or (ii) when pregnancies are discovered after performing the examinations. However, examinations are documented in those cases where the medical physicists are notified and asked to evaluate the dose and assess the risk to the foetus.

Material and methods

The survey method

- As a first approach, a survey method was designed. Forms were sent to diagnostic radiology departments in 3 counties of Sweden to gather information about examinations performed during pregnancy. The forms were handled by the head nurses at each radiology department, and fill-in instruction were given to their staff. The following information was requested;
- routines used to avoid unintentional exposures of foetus,
- medical indication and diagnostic method
- patient data: ID, date of birth, length, weight, gestational age
- exposure data: (i) X-ray diagnostics: X-ray equipment, no of projections, number of exposures, tube potential, mAs, fluoroscopy duration, field size, focus-skin distance etc., (ii) CT-diagnostics: tube potental, mAs/slice, slice thickness, number of slices, etc, (iii) nuclear medicine diagnostics: type of radiopharmaceutical, administered activity, hydration, blocking agents etc.

The survey was run for 12 month, from April 1999 to March 2000. Results obtained from the first quarter showed a very low reporting frequency, and the head nurses were reminded of the reporting request. For the full study period about 28 000 examinations of women aged 15-50 years were performed, but only 21 cases were reported where the patient was pregnant at the time of examination, which was far less than anticipated. Therefore alternative methods to identify the true frequency was adopted.

The patient registry method

In Sweden the authorities keep population registries of all individuals residing in Sweden. Such records are much used in public health-studies, both for prospective and retrospective studies. Hence, the Swedish hospitals have access to some of these registries and we therefore wanted to explore the possibility to retrospectively obtain data on patients exposed during pregnancies. In order to use the registries and produce files of patient ID-records clearance had to be obtained from both the local hospital registry boards and from the ethical committee at the Linköping University. Data were then extracted from the hospital RIS (Radiological Information Systems) files used for registration of radiation diagnostic procedures. For X-ray examinations patient ID, date of examination, diagnostic procedure (type and location) was extracted for a 8 months time period. For nuclear medicine imaging, where patient flow is much smaller than for X-ray diagnostics, the corresponding data was extracted for a 5-year period (1995-1999).

The data search was then performed according to the following search routine:

1 Extract patient data from the RIS for a given period and export to PC-Microsoft Excel;

- patient ID (name, date of birth[1])
- date and exam type and code (conventional X-ray, CT, nuclear medicine imaging, MR, ultrasound)

2 Extract data by gender and age; female aged 15-50 years

3 Export date of birth to text-file, send data to the National Tax Board (RSV)

4 Data generated by RSV and sent to customer (text-file on diskette):

- date of birth for patient and if applicable their children
- if applicable date of emigration
- if applicable date of death

5 RSV data imported in PC-Microsoft Excel and data then sorted and extracted;

- keep only data with patient and children
- extract data including children birth dates within the study period plus 9 months.

6 Manually extract data where patient were pregnant at time of examination

7 Sort selection by type of examination

8 Calculate foetus age at time of examination

[1]the birth data also include a 4 digit number (Swedish social number system), together giving a 12 digit number unique for the person. Thus, name of patient not needed for the search.

Results obtained by the patient registry method

X-ray diagnostic procedures

Figure 1 shows the number of pregnant patients examined by different X-ray diagnostic procedures during the 8 months study period. Out of about 19 000 X-ray examinations 299 cases of foetus exposures were identified of which 82 cases at the Linköping Univ Hospital, 108 cases at the County Hospital Ryhov, and 109 cases at the Örebro Univ Hospital. Thus for this patient group about 1.6% were pregnant at the time of X-ray examination. As expected, the pelvis examinations predominate followed by examinations of extremities, which is the most common X-ray examination. Also the relatively high frequency of chest/pulm examinations is expected since chest/pulm problems may occur during pregnancies.

Only a few of the examination types may result in significant radiation dose to the foetus. Helmrot *et. al* (2002) has shown that examinations where the radiation field does not include the uterus give only marginal doses to the foetus. These include many of the examinations found in this study, i.e. extremities, CT skull/brain, cervical spine, sinus/esophagus/hypopharynx, flebography of leg, mammography, odont panorama, thoraic spine. However, for the remaining examination types the uterus is partly or completely included in the radiation field. In the case of chest and pulm investigations the fraction of uterus exposed highly depends on the gestational age, being insignificant in early pregnancy.

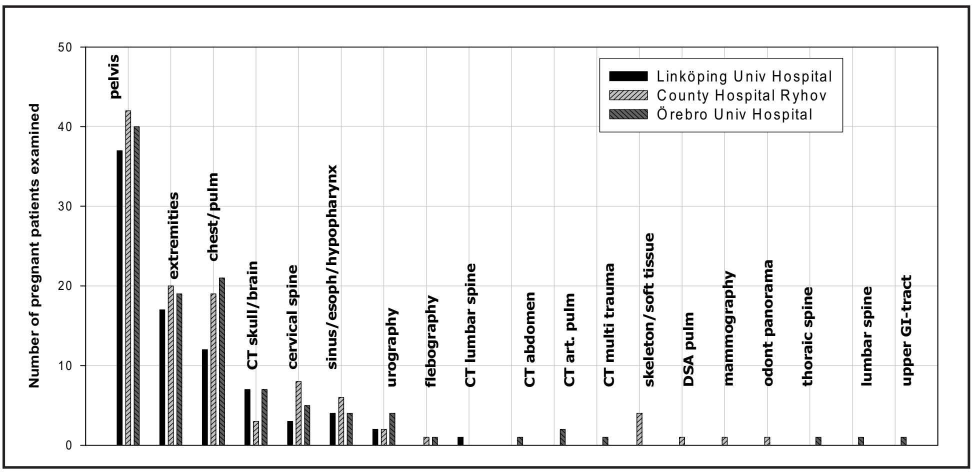

Figure 1. Number of pregnant patients examined by X-ray diagnostic procedures during a 8 months study period. The total number of examinations of female patients were 6697, 5814 and 6204 respectively for the 3 hospitals (Linköping Univ Hospital, County Hospital Ryhov and Örebro Univ Hospital).

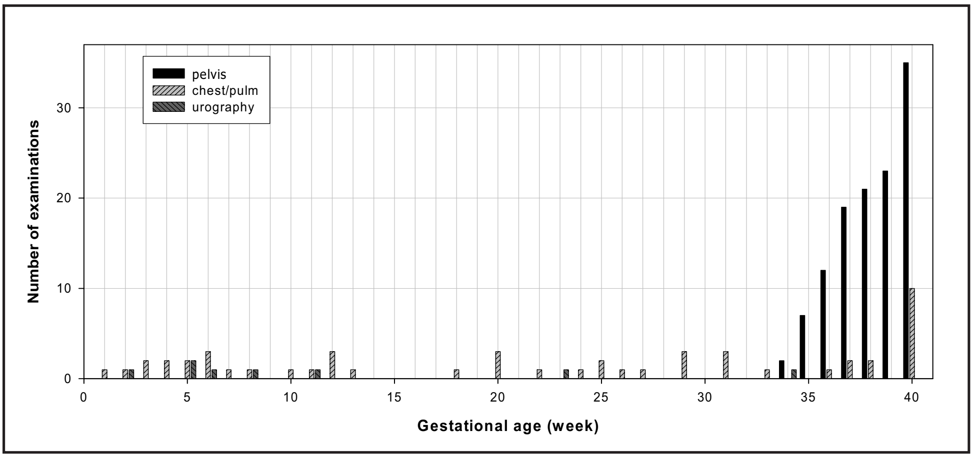

Figure 2. Number of pregnant patients examined by X-ray diagnostic procedures during a 8 months study period versus gestational age. Examinations performed at Linköping Univ Hospital, County Hospital Ryhov and Örebro Univ Hospital.

Figure 2 shows the number of pelvis, chest/pulm and urography examinations versus gestational age. As expected the pelvis investigations are performed close to the time of delivery. However, the chest/pulm and urography investigations are more evenly spaced in time, which was the case also for most of the other examination types. It could be concluded that most of the investigations were carried out on known pregnancies since very few investigations were made during the first few weeks after conception when pregnancy might not be known.

Nuclear medicine imaging procedures

Figure 3a shows the number of pregnant patients examined by nuclear medicine imaging procedures at the Linköping Univ Hospital during the years 1995 to 1999. The Figure also shows the same data given as the fraction (%) of the total number of examinations. Out of 4507 examinations 33 cases of pregnant patient exposure were identified. The predominant investigation is PMP Pulm Perfusion using ^{99m}Tc. Such investigations, together with PMV Ventilation Study, are expected since lung embolus is not an uncommon complication during pregnancies. As shown in Figure 3b, the PMP investigations are quite evenly spaced in gestational age.

Discussion and Conclusions

By using existing radiological information systems (RIS) together with national population records it has been shown that thorough information about

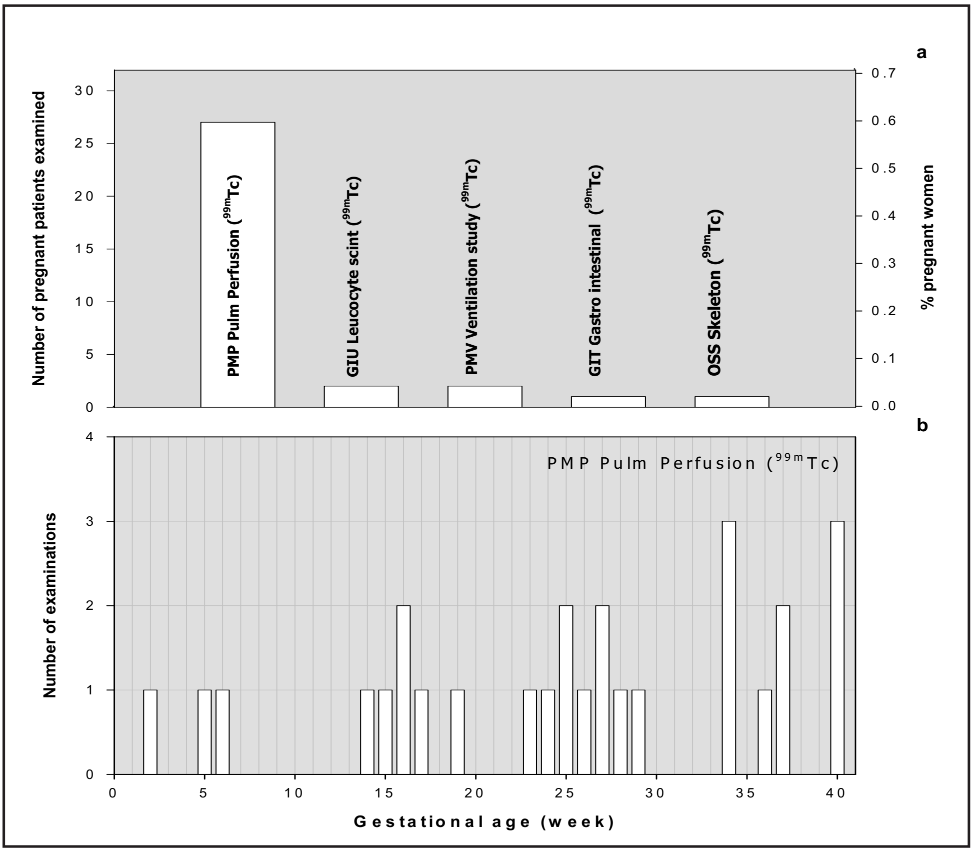

Figure 3. a) Number of pregnant patients examined by nuclear medicine imaging during a 5 year study period (1995-1999). Also shown the fraction (%) of patients examined that were pregnant at the time of examination (in relation to the total number of examinations of female patients aged 15-50 y (4507 exams). b) Number of pregnant patients examined versus gestational age.

both frequency and type of radiation diagnostic procedures performed on pregnant patients can be obtained. Also the gestational age at exposure could be determined. To obtain these data by query form surveys proved to be unsuccessful. However, if needed, it could probably be improved in the future by supplying pregnancy information directly into RIS.

The study has shown that, apart from pelvis investigations, few of the X-day examinations result in significant radiation dose to the foetus since the radiation field does not include the uterus. However, in the case of chest and pulm investigations the dose highly depends on gestational age, with doses increasing with gestational age. In the case of nuclear medicine imaging, studies of lung perfusion/ventilation predominate for pregnant patients. The time of examinations are distributed among all gestational age.

The frequency of examinations on pregnant patient aged 15-50 years were 1.6% and 0.7% for X-ray and nuclear medicine imaging investigations, respectively. For this age group, in the general population about 6% are expected to be pregnant, suggesting that for patients of known pregnancies referral to alternative non-radiation diagnostic procedures is common and/or the pregnant population are in less demand of radiation diagnostic procedures.

The use of the patient registry method has great potential for large-scale studies, i.e. national studies of the same kind as this study, but could also be used for retrospective health studies. For example, studies of low dose cancer incidence among people exposed as foetus could be performed. However, large studies would require automatic computer routines/programmes to speed up data searching and extraction.

Acknowledgements

This work was supported by The Swedish Radiation Protection Authority grant SSI P 1114.98.

References

[1] Committee on the Biological Effects of Ionizing Radiaitons. Health effects of exposure to low levels of ionizing radiation. BEIR V. Washington DC: National Academy Press, 1990.

[2] Helmrot E, Pettersson H, Sandborg M, Olsson S, Nilsson J, Cederlund T. Radiation doses to the unborn child at diagnostic examinations in Sweden. Paper presented at the European IRPA Congress, Florence, 8-11 Oct 2002.

[3] ICRP Publication 84. Pregnancy and Medical Radiation. Annals of the ICRP 2000; 30/1.

7i Doskatalogen på internet

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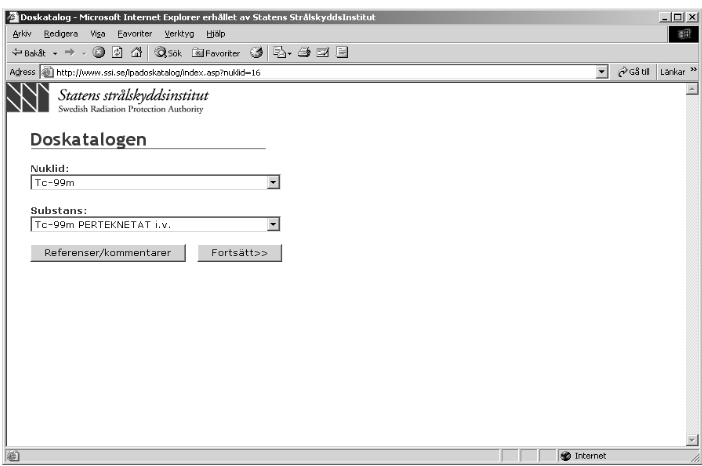
Statens strålskyddsinstitut, SSI, har under ett antal år stött arbetet kring förbättrade stråldosuppskattningar till patienter vid nukleärmedicinska undersökningar. Projektets namn var från början "Stråldoser från radioaktiva ämnen i medicinskt bruk", men har mer och mer gått under benämningen "Doskatalogen". Arbetet började redan 1969 och stödet har lett till ett mångårigt engagemang inom ICRP:s "Task Group on Dose to Patients from Radiopharmaceuticals".

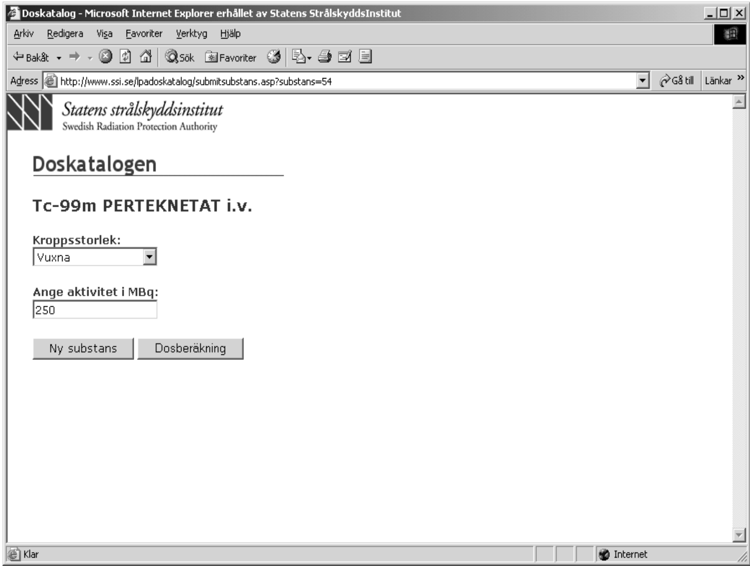
Doskatalogen är en databas över stråldoser från radiofarmaka som används inom nukleärmedicin och är avsedd att användas av strålskyddsexperter inom sjukvård och forskning. Doskatalogen används frekvent inom den svenska sjukvården och består av fler än 190 olika substanser och radionuklider som presenterats i pappersformat. Nu kan även stråldoser från ett 60-tal av de vanligaste radiofarmaka beräknas via SSI:s hemsida, www.ssi.se under rubriken "Sjukvård". Detta gör Doskatalogen mer lättåtkomlig för alla som har behov utav den.

I exemplet nedan görs en stråldosberäkning för en vuxen person som injicerats med 250 MBq Tc-99m perteknetat.

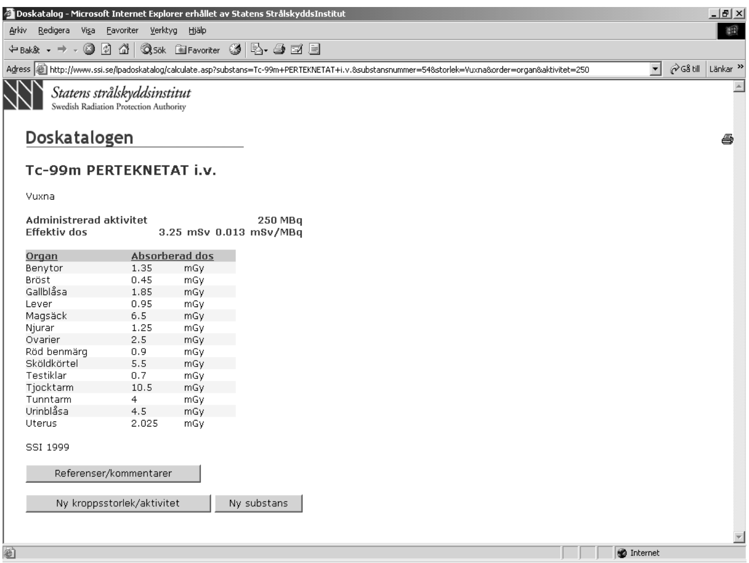
1) Välj radionuklid och substans. Genom att välja en radionuklid ses enbart substanserna med den nukliden. Då "alla" är vald för nuklider så ses alla substanser. Kommentarer och referenser till de dosfaktorer som används kan ses under "Referenser/kommentarer".



2) Välj vilken kroppsstorlek som beräkningen ska göras med, de vanligaste kroppsstorlekarna är vuxen, 15, 10, 5 och 1 år, men för vissa radiofarmaka finns det även beräkningar för andra kroppsstorlekar. Ange också vilken aktivitetsmängd i MBq som använts.



3) Vid dosberäkningen fås organdoserna och den effektiva dosen. Resultatet kan sorteras i bokstavsordning alternativt i storleksordning på stråldosen. SSI XXXX anger det årtal substansen publicerades inom Doskatalogen.



Följande substanser som finns inlagda i doskatalogen på SSI:s hemsida.

Ba-133m BARIUMJON i.v.
C-11 ERYTROCYTER i.v.
C-11 KOLMONOXID inh
C-11 TYMIDIN [2-11C] i.v.
C-11 TYMIDIN [Metyl-11C] i.v.
C-14 FETTSYROR i.v.
C-14 UREA p.o.
Co-57 VITAMIN B12 i.v. (utan bärare)
Co-57 VITAMIN B12 p.o.
Co-57 VITAMIN B12 Schillingtest p.o.
Co-58 VITAMIN B12 i.v. (utan bärare)
Co-58 VITAMIN B12 p.o.
Co-58 VITAMIN B12 Schillingtest p.o.
Cr 51 LEUKOCYTER i.v.
Cr 51 TROMBOCYTER i.v.
Cr-51 ERYTROCYTER i.v.
Cr-51 KROM-EDTAT i.v.
Cu-64 KOPPARJON i.v.
Cu-64 KOPPARJON p.o.
F-18 FDG i.v.
Ga-67 GALLIUMCITRAT i.v.
H-3 FETTSYROR i.v.
I-123 AMFETAMIN i.v.
I-123 MIAA i.v.
I-123 MIBG i.v.
I-123 NATRIUM-o-JODOHIPPURAT i.v.
I-131 ALBUMIN i.v.
I-131 JODIDJON (blockerad thyreoidea) i.v.
I-131 JODIDJON i.v & p.o.
I-131 JODMETYLNORKOLESTEROL i.v.
I-131 MIAA i.v.
I-131 MIBG i.v.
I-131 NATRIUM-o-JODOHIPPURAT i.v.
In-111 HUMANT IMMUNOGLOBULIN i.v.
In-111 OCTREOTID i.v.
O-15 VATTEN i.v.
Se-75 GALLSYRA i.v.
Tc-99m ALBUMIN MIKROSFÄRER i.v.
Tc-99m DMSA i.v.
Tc-99m DTPA i.v.
Tc-99m EC (akut ensidig njurblockering) i.v.
Tc-99m EC (nedsatt njurfunktion) i.v.
Tc-99m EC (normal njurfunktion) i.v.
Tc-99m ECD i.v.
Tc-99m ERYTROCYTER i.v.
Tc-99m FOSFATER OCH FOSFONATER i.v.
Tc-99m FURIFOSMIN (Q12) (arbete) i.v.
Tc-99m FURIFOSMIN (Q12) (vila) i.v.
Tc-99m HIG i.v.
Tc-99m HMPAO i.v.
Tc-99m LEUKOCYTER i.v.
Tc-99m LEVER-GALLVÄGSSUBSTANSER i.v.
Tc-99m MAA i.v.
Tc-99m MAG-3 i.v.
Tc-99m OLÖSLIGA MARKÖRER (fast) p.o.
Tc-99m OLÖSLIGA MARKÖRER (flytande) p.o.
Tc-99m PERTEKNEGAS inh.
Tc-99m PERTEKNETAT (Blockering) i.v.
Tc-99m PERTEKNETAT i.v.
Tc-99m PERTEKNETAT p.o.
Tc-99m SMÅ KOLLOIDER i.v
Tc-99m STORA KOLLOIDER i.v.
Tc-99m TEKNEGAS inh.
Tc-99m TETROFOSMIN (arbete) i.v.
Tc-99m TETROFOSMIN (vila) i.v.
Tl-201 TALLIUMJON i.v.
Xe-133 KOKSALTLÖSNING i.v.
Xe-133 XENONGAS inh.

7j The nuclear medical diagnostic procedures at the hospitals in Sofia over a period of ten years – from 1990 to 2000 – structure, number and doses

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Abstract

The aim of this study is to analyze the structure and the number of nuclear diagnostic procedures at hospitals in Sofia over a period of ten years (from 1990 to 2000); to calculate the effective doses received by patients; to compare the results from different hospitals.

We have developed a form in order to collect the necessary information from hospitals /nuclear laboratories/. The information has been collected according to examined organ, applied radioactive pharmaceutical and injected activity [MBq], patient's age and sex.

The effective doses have been calculated using transition coefficients /1/ for each of the applied radioactive pharmaceuticals and for each of the diagnostic procedures.

We have received the information and calculated the doses. We have compared the data from different hospitals /nuclear laboratories/. The results show that the doctors in different hospitals have used different radioactive pharmaceuticals and applied different activities for one and the same procedure.

Keywords

nuclear medicine, low doses radiation, dosimetry, effective dose

Introduction

Radiation exposure of Bulgarian population is due to different sources /table 1/ /2/. The largest contributions are natural background radiation (about 2/3 of the total) and exposures from medical diagnosis (about ¼ of the total).

Table 1. Distribution of overall exposure by 1990 years.

Source of exposure	Mean effective doses [μSv/a]	Collective doses [man-Sv/a]	% of the total exposure
1. Background exposure	**2283**	**19600**	**68,30**
External	735	6300	22,00
Internal	1548	13300	46,30
2. Excess over background exposure	**1054**	**9100**	**31,70**
Occupational exposure	116	1000	3,50
X-ray diagnosis	*802*	*6900*	*24,00*
Radioisotope diagnosis	*81*	*700*	*2,40*
Uranium mining	41	350	1,20
NPP	3	22	0,08
TPP	3	23	0,08
Global fallout	2	15	0,05
Other sources	6	51	0,18
TOTAL EXPOSURE	**3337**	**28700**	**100**

Table 2 shows data of exposure of Bulgarian population due to radioisotope diagnosis /3/.

Table 2. Exposure of Bulgarian population due to radioisotope diagnosis.

Year	Number of examinations per year		Mean effective dose [mSv/a]		Collective effective dose [man-Sv/a]
	total /thousands	per capita	per examination	per capita	
1970	34	0,004	17,0	0,07	0,6
1975	70	0,008	13,0	0,10	0,9
1980	116,4	0,013	8,8	0,115	1,022
1985	170	0,020	7,0	0,13	1,2
1990	130	0,015	5,0	0,08	0,7
1992	85	0,010	5,0	0,05	0,4
1993	89	0,011	5,3	0,05	0,4

The aim of the study

The aim of this study is to analyze the structure and the number of nuclear medical diagnostic procedures at hospitals in Sofia from 1990 to 2000; to calculate the effective doses received by patients; to compare the results from different hospitals.

Materials and Methods

We have developed a form /table 3/ for collecting the necessary information from hospitals - nuclear laboratories.

Table 3. Nuclear diagnostic procedures.

Exam-ined organ	Number of proce-dures annually	Radio active pharma ceutical	Prescribed activity [MBq]	Age and sex distribution (number)											
				0-1 year		2-5 years		6-10 years		11-15 years		16-40 years		>40 years	
				m	f	m	f	m	f	m	f	m	f	m	f

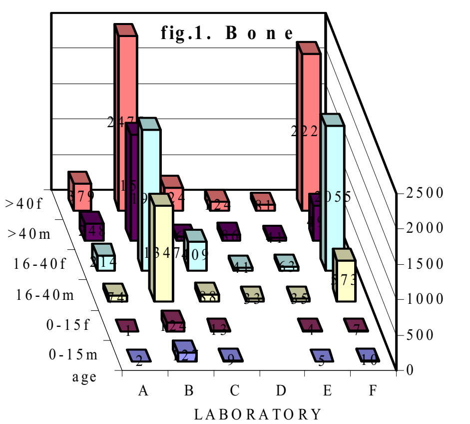
We have taken “Form 3: Nuclear medicine procedures” – Questionnaire for a Global Survey of Medical Radiation Usage and Exposure from 1995 to the present – UNSCEAR /4/ as a pattern and we have included additional columns. The information has been collected according to the examined organ: bone, heart, lung, thyroid, kidney, liver, brain and other; applied radioactive pharmaceuticals and injected activities [MBq]; patient’s age and sex.

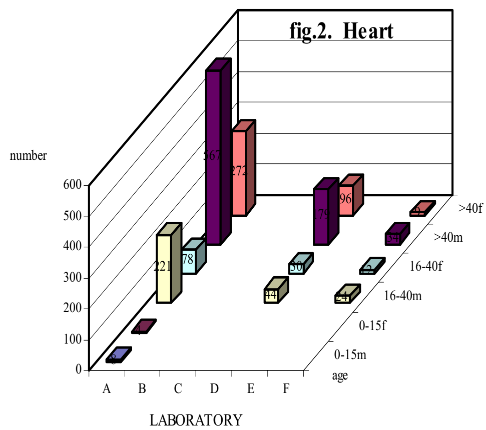
The effective doses have been calculated using transition coefficients /1/ for each of the applied radioactive pharmaceuticals and for each of the diagnostic procedures.

We have taken the “administered activity” [MBq] and the respective “transition coefficients” [mSv/MBq] to calculate the effective dose [mSv] – mean and variation. We have compared the results from different hospitals and we present them on figures and in tables.

Results

A: Number of patients and structure of the nuclear diagnostic procedures /fig1-fig8/





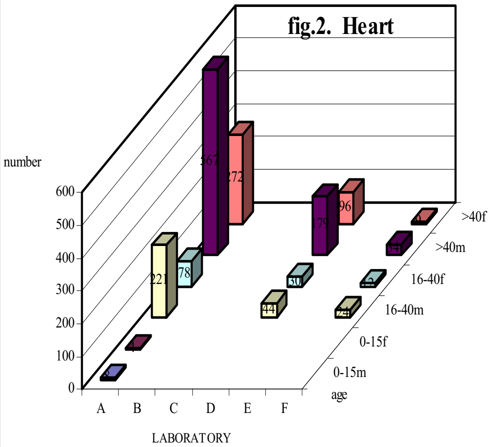


fig.2. Heart
number
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500
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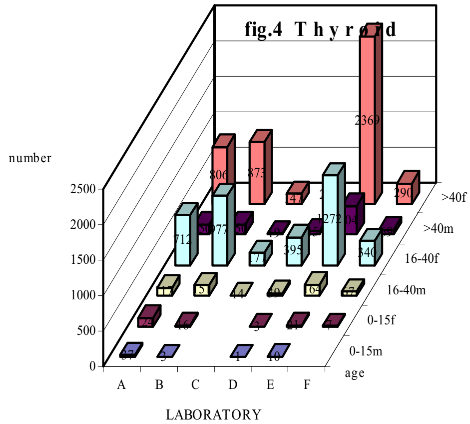
fig.4 T h y r o i d
number
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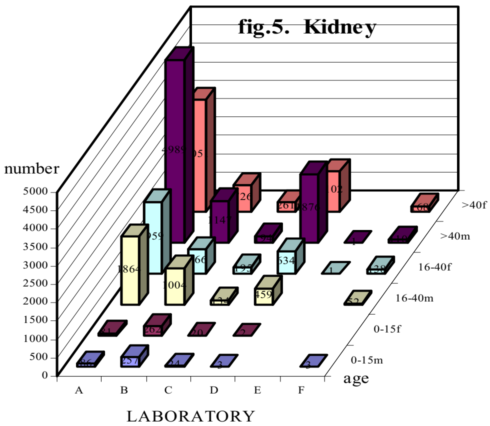
fig.5. Kidney
number
5000
4500
4000
3500
3000
2500
2000
1500
1000
500
0
A B C D E F
LABORATORY
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>40m
16-40f
16-40m
0-15f
0-15m
age

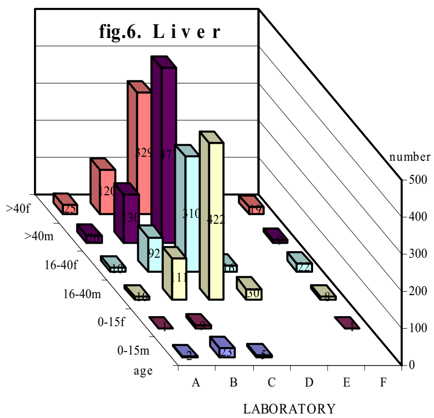
fig.6. L i v e r
number
500
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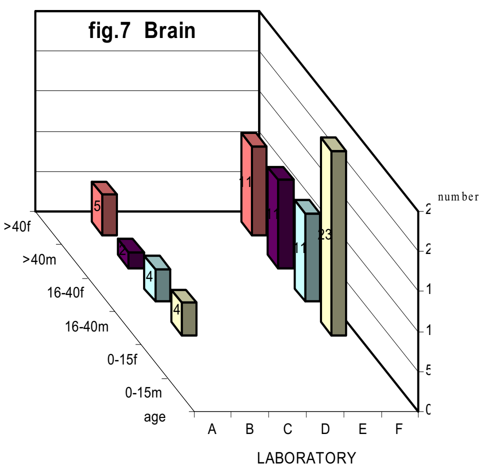
fig.7 Brain
number
>40f
>40m
16-40f
16-40m
0-15f
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age
A B C D E F
LABORATORY

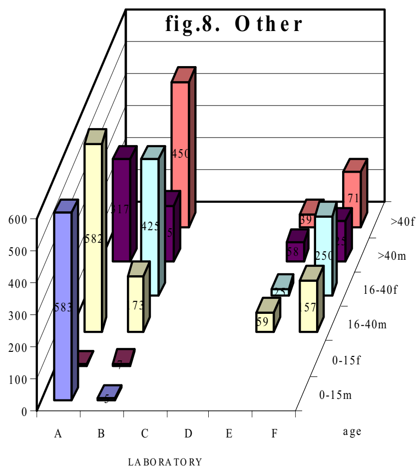
fig.8. O t h e r
600
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A B C D E F
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The figures show that there are high variations in number of patients and structure of nuclear diagnostic procedures in different hospitals in Sofia.

B: The doctors from different hospitals have applied different radioactive pharmaceuticals and they have prescribed different activities. The tables /from 4 to table 11/ show the effective doses [mSv] received by patients at different hospitals /laboratories: A, B, C, D, E and F/ for each of the diagnostic procedures according to examined organ: bone, heart, lung, thyroid, kidney, liver, brain and other.

Table 4. Effective doses received by patients from bone examinations.

Radioactive pharma-ceutical	Lab. A eff. doses [mSv]	Lab. B eff. doses [mSv]	Lab. C eff. doses [mSv]	Lab. D eff. doses [mSv]	Lab. E eff. doses [mSv]	Lab. F eff. doses [mSv]	Min and max eff. doses [mSv]
Ga Citrat						11,10 - 13,32	11,10 - 13,32
Tc HDP		2,38-4,49			3,37 - 5,13		2,38 - 5,13
Tc MDP	1,76-2,29	2,00-4,58	0,9 - 1,6	1,76 - 2,29	3,27 - 3,44	0,65 - 4,58	0,65 - 4,58
Tc MIBI					6,60 - 8,16	4,88 - 10,88	4,88 - 10,88
NaI					225,0 - 136,0	224,96 - 3404	136,0 - 3404,0
Tc PYP	1,83-2,33						1,83 - 2,33
Tc Chlorid						122,10 - 202,8	122,1 - 202,76

Table 5. Effective doses received by patients from heart examinations.

Radio-active pharma-ceutical	Lab. A eff. doses [mSv]	Lab. B eff. doses [mSv]	Lab. C eff. doses [mSv]	Lab. D eff. doses [mSv]	Lab. E eff. doses [mSv]	Lab. F eff. doses [mSv]	Min and max eff. doses [mSv]
Tc MIBI	3,70-8,16						3,70 - 8,16
Tc MIBI rest				4,92 - 6,03			4,92 - 6,03
Tc MIBI stres				3,96 - 4,85			3,96 - 4,85
Tc Chlorid						12,21 - 202,76	12,21 - 202,76

Table 6. Effective doses received by patients from lung examinations.

Radio-active pharma-ceutical	Lab. A eff. doses [mSv]	Lab. B eff. doses [mSv]	Lab. C eff. doses [mSv]	Lab. D eff. doses [mSv]	Lab. E eff. doses [mSv]	Lab. F eff. doses [mSv]	Min and max eff. doses [mSv]
Tc MAA	0,92 -1,71	0,82 - 2,85					0,82 - 2,85
Tc MIBI rest		9,84 -12,06				1,44 - 12,03	1,44 - 12,06
Ga Citrat						11,10 - 13,30	11,10 - 13,33
Tc HSA[a]		1,48					
a – we have not had the transition coefficients for this radiopharmaceutical for each of the age groups so we have not calculated the variations of effective dose							

Table 7. Effective doses received by patients from thyroid examinations

Radioactive pharma-ceutical	Lab. A eff. doses [mSv]	Lab. B eff. doses [mSv]	Lab. C eff. doses [mSv]	Lab. D eff. doses [mSv]	Lab. E eff. doses [mSv]	Lab. F eff. doses [mSv]	Min and max eff. doses [mSv]
NaI	8,24 - 19,04				34,04 - 87,14	28,12 - 8288	8,24 - 8288
Tc Perteh		1,03 - 1,27	0,60 -0,80	1,66 -2,07	0,84 - 1,15	0,58 - 7,25	0,58 - 7,25
Tc MIBI re		13616 - 16576			7,38 - 9,38		7,38 - 16576

Table 8. Effective doses received by patients from kidney examinations.

Radioactive pharma-ceutical	Lab. A eff. doses [mSv]	Lab. B eff. doses [mSv]	Lab. C eff. doses [mSv]	Lab. D eff. doses [mSv]	Lab. E eff. doses [mSv]	Lab. F eff. doses [mSv]	Min and max eff. doses [mSv]
I Hipur	0,05 - 0,13			12,32 -12,99		4,95 - 8,66	0,05 - 12,99
Tc DTPA	2,62 - 3,57	0,96 - 3,48				0,33 - 2,15	0,33 - 3,57
Tc DMSA	0,59 - 0,99	0,55 - 1,19	0,10 - 0,28	1,12 - 2,38	5,06 - 5,94		0,10 - 5,94
Tc MAG_3	0,39 - 1,55	1,77 - 5,18		3,70 - 5,18			0,39 - 5,18
Tc EC[a]	1,00						1,00
a – we have not had the transition coefficients for this radioactive pharmaceutical for each of the age groups so we have not calculated the variations of effective dose							

Table 9. Effective doses received by patients from liver examinations.

Radioactive pharma-ceutical	Lab. A eff. doses [mSv]	Lab. B eff. doses [mSv]	Lab. C eff. doses [mSv]	Lab. D eff. doses [mSv]	Lab. E eff. doses [mSv]	Lab. F eff. doses [mSv]	Min and max eff. doses [mSv]
Tc SC nor	0,03 - 0,11	0,86 -1,91	0,11 - 0,31			0,59 - 2,29	0,03 - 2,29
Tc IDA	0,45 - 2,51	1,15 -3,98					0,45 - 3,98
Tc RBC		3,32 -4,21				1,45	1,45 - 4,21
Tc Fitat[a]							
Tc Pertehnet.		2,68 -3,29					2,68 - 3,29
a – we have not had the transition coefficients for this radioactive pharmaceutical for each of the age groups so we have not calculated the variations of effective dose							

Table 10. Effective doses received by patients from brain examinations.

Radio-active pharma-ceutical	Lab. A eff. doses [mSv]	Lab. B eff. doses [mSv]	Lab. C eff. doses [mSv]	Lab. D eff. doses [mSv]	Lab. E eff. doses [mSv]	Lab. F eff. doses [mSv]	Min and max eff. doses [mSv]
Tc HMPA						2,22-12,58	2,22- 12,58
Tc DTPA		3,67-5,00					3,67-5,0
Tc HSA						1,15-1,40	1,15-1,4
Tc Perteh-netat		5,90-7,25					5,90-7,25

Table 11. Effective doses received by patients from other procedures.
/testis, gl.salivares, gl.mammae, melanoma malig., NHL, NL, articulations, lyonf., ISG /

Radioactive pharmaceu-tical	Lab. A eff. doses [mSv]	Lab. B eff. doses [mSv]	Lab. C eff. doses [mSv]	Lab. D eff. doses [mSv]	Lab. E eff. doses [mSv]	Lab. F eff. doses [mSv]	Min and max eff. doses [mSv]
1.Tc Pertehnetat	5,20-11,40						5,20-11,40
2.Tc MIBI rest		9,84-12,06					9,84-12,06
Tc Pertehnetat		0,87-1,55					0,87-1,55
3.Tc MIBI		9,84-12,06				7,38-12,06	7,38-12,06
4.Tc MIBI						2,66-8,16	2,66-8,16
5.Ga Citrat						14,80-17,76	14,80-17,76
6.Tc Albumin		2,90-3,54					2,90-3,54
Tc SC nor		1,61-2,06					1,61-2,06
Tc MIBI rest		9,84-12,06					9,84-12,06
Tc NanoColoid					0,11-0,13		0,11-0,13
Tc HAS					0,05-0,06		0,05-0,06
7.Tc Pertehnetat		1,52-1,85					1,52-1,85
8.Tc NanoColoid						1,07-13,1	1,07-13,1
9.Tc HSA						2,30-4,74	2,30-4,74
Tc MIBI						2,22-10,88	2,22-10,88
Tc Chlorid						12,21-20,30	12,21-20,3

Conclusions

1. In different hospitals there are high variations in number of patients and structure of nuclear diagnostic procedures;

2. The doctors in different hospitals apply different radioactive pharmaceuticals and they prescribe different activities for one and the same procedure;

3. In different hospitals there are high variations in effective doses [mSv].

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7k Nuclear medicine practices in Sweden – the development during the last 30 years

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Introduction

Clinics with nuclear medicine practices are obliged since 30 years ago to report annually to the Swedish Radiation Protection Authority (SSI) the number of examinations and treatments performed, including the administered activities. SSI is putting together these data and these compilations are used for documentations of statistics and following trends for nuclear medicine examinations and treatments.

Material

SSI's compilations of the reports from the clinics are a basis for the calculation of radiation doses to individuals for the various examinations, and also for the calculation of the collective dose. The compilations are containing data such as type of radio pharmaceutical, the mode of administration, number of hospitals, total number of examinations and treatments, mean value of the administered activity, the interval of mean values for administered activity for the various hospitals and the maximum activity for the individual patient. The trends for the past years are presented and the coming development with new challenges is discussed.

Results

The total number of examinations has doubled since 1968. 1968 the number was 48 000, today approximately 102 000 examinations are performed. Figure 1 is giving the total number of nuclear medicine examinations performed in Sweden between 1968-2001.

The type of radionuclides for the examinations has changed considerably. In the 1960s iodine isotopes were used frequently, which is seen in the figure below. In the 1970s Tc-99m became more and more dominant. In 2001 80 000 nuclear medicine examinations were performed with Tc-99m.

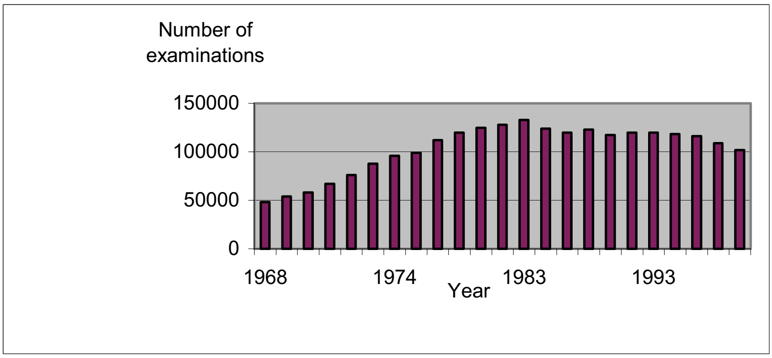

Figure 1. Total number of nuclear medicine examinations in Sweden 1968-2001 (all types of *in vivo* examinations are included).

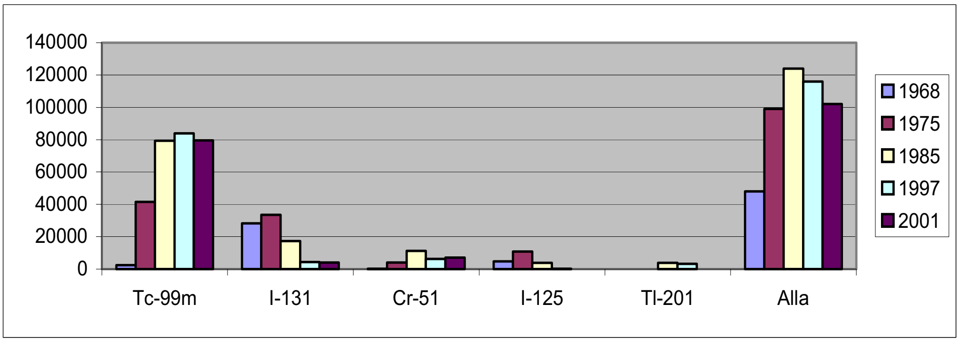

Figure 2. Number of nuclear medicine examinations for the most common radionuclides.

The distribution of types of examinations has also been changed during the years. This is demonstrated in the table below (table 1). The changes depend among other things on the development of alternative diagnostic methods and on the technical progress concerning equipment but also on the development of new radioactive drugs.

Table 1. Number of examinations for various types of examinations expressed as percentage of the total number of examinations.

Year	Bone scint. Tc-99m	Tyroid scint. I-131	Brain scint. Tc-99m	Liver scint. Tc-99m	Lung scint. Tc-99m	Kidney clear. Cr-51	Heartscint. Tl-201
1968	-	25	2	1	0,1	0,1	-
1975	6	8	13	13	3	3	0,1
1985	28	4	4	6	8	9	2
1997	26	0,6	0,4	0,2	11	5	3
2001	24	0,1	1	0,1	7	6	0,8

Discussion

The collective dose to patients has been reduced by a factor of two. This is because of the drastic reduction of the use of iodine isotopes being substituted by Tc 99m. Also other relatively long-lived isotopes have been replaced by Tc 99m.

Other methods, for instance x-ray examinations, have replaced the use of radionuclides for examination of the liver and the brain. The number of these examinations has decreased significantly.

There are three examinations that provide the main contribution to the collective dose: bone-, lung- and kidney-scintigraphy. One of the new challenges in nuclear medicine is the development of radio nuclides and examination methods for PET (**p**ositron **e**mission **t**omography) work. A distinctive trait for PET radio nuclides is their short half-life. This implies that high activities are handled and administered to the patient which is putting high demands on radiation protection measures. But the expansion of this method is impeded in Sweden (as well as in other countries) by the fact that the production costs are high and that the transportation of the radio nuclides to other hospitals can have problems.

The nuclear medicine gamma cameras have become more sophisticated and effective. For instance with a three-headed gamma camera the administered activity can be reduced and/or the image quality increased or the examination time reduced. Gamma cameras combined with Computed Tomography allow a detailed anatomical allocation

7l Radiation dose to adult patients in LS spine X-ray examinations of health centres in one central hospital district in Finland

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Introduction

According to the Medical Exposure Directive (97/43/Euratom)[1], the radiation dose to the patient should be measured and the doses compared against to the national reference doses. The European Commission has issued quality criteria for x-ray examinations for adults[2]. The criteria to be considered include clinical image quality, examination techniques and radiation dose. The radiation dose caused by x-ray examinations to adult patients should be measured or calculated for at least ten standard size patients (70 kg ± 15 kg). At every unit of the department, the radiation dose is determined, at regular intervals, for the most general examinations, at least in one projection. If the comparison dose levels are exceeded repeatedly, both the examination techniques and the radiological units in use must be checked and any corrective actions required must be carried out.

The Nordic countries defined Nordic reference dose levels for some X-ray examinations in 1996[3]. In Finland, dose reference levels for adults were given on 8 December 2000[4]. Table 1 shows the national dose reference levels for LS spine examinations in Finland, in the EU and in the Nordic countries.

Table 1. Dose reference levels of the EU, the Nordic countries and Finland as the surface dose (ESD) and as the dose-area product (DAP) for adult LS spine examinations.

Dose quantity	EU AP LAT LSJ	Nordic countries AP LAT LSJ	Finland AP LAT LSJ
ESD (mGy)	10 30 40	6 - -	8 25 35
DAP (Gy*cm ^{2})	-	10 (AP+LAT)	10 (AP+LAT)

Statistics on x-ray examinations in health centres

Health centres play an important role in radiology in Finland. A total of 220 health centres perform approximately 37 % of all x-ray examinations[5]. The 13 health centres located in the central hospital district selected for this study carried out about 67500 x-ray examinations in 2000. Of these, 3200 were LS spine examinations. LS spine examinations accounted for about 5 % of all x-ray examinations. Figure 1 shows some statistics on radiological examinations in the health centres included in this study.

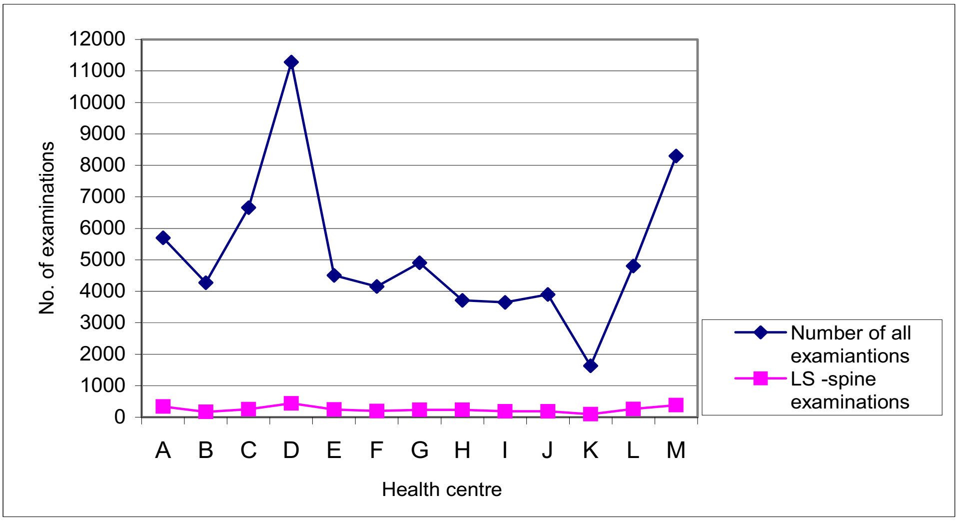

Figure 1. Statistics on radiological examinations in the health centres of this study. Number of all examinations and LS-spine examinations.

Material and methods

The radiation doses of LS spine examinations in13 health were measured for 249 patients in the AP and LAT projections and for 97 patients in the LSJ projection. In all of the health centres, the LS spine examination included 2.4 X-ray films on average. The dose area product (DAP) was measured with a DAP meter and the entrance surface dose (ESD) was calculated on the basis of the X-ray tube output, the examination techniques and patient data (height, weight). The average patient weight varied from one health centre to another from 68.2 kg to 73.7 kg the average weight of all patients being 71.6 kg. The organ dose (bone marrow, gonades) and the effective dose of various projections were calculated by using the PCXMC[6] dose calculation program. Radiographers recorded the patient data, examination techniques and dose data on questionnaires. The DAP monitors were calibrated separately in all of the health centres before the measurements by comparing the DAP monitors against the calibrated radiation meter of MDH in a secondary standard laboratory. The examination techniques were compared against European quality criteria and the radiation doses against the national and European reference levels.

Results

Examination techniques

Table 2 shows the examination techniques used in various health centres for LS spine examinations.

Table 2. Examination techniques used in various health centres in LS-spine x-ray examinations.

Health centre	Generator	Unit	Nominal focal spot	Total filtration	Grid	Speed class of image receptor			FFD	AEC
			(mm)	(mmAl)		AP	Lat	LSJ	(cm)	
E	Finnorgan	Multix	1.2	3.5	12:1/40	400	adual	400	115	+
H	ShimazuUD	Philips	2.0	3.0	--	400	400	200	115	+
L	ShimazuUD	RS85	1.2	3.0	8:1/40	200	400	400	114	+
F	Medio 50	Philips.diag	1.2	4.0	12:1/36	400	400	250	110	+
B	BR-2002	BR-1700	2.0	5.0	8:1/40	400	Gradual	250	110	+
A	Optimus 50	Bucky diag	1.0	4.8	8:1/36	200	200	800	110	+
M	BR-1001	BR-1700	1.8	5.0	8:1/34	400	Gradual	700	110	+
D	Diagnost 96	Philips.diag	1.3	3.2	13:1/60	250	250	250	110	+
G	BR-2012	BR-1700	2.0	5.0	10:1/40	200	Gradual	200	110	+
C	BR-1001	Philips	1.8	3.0	8:1/36	Digi	Digi	Digi	115	+
K	BR-2012	BR-1700	2.0	4.0	8:1/40	400	Gradual	400	110	+
J	BR-2012	BR-1712	2.0	2.7	10:1/40	400	Gradual	400	110	+
I	BR-2012	BR-1700	2.0	5.5	- -	200	700	200	110	+

The tube voltage in the AP projection ranged from 65 kV to 85 kV, in the LAT projection from 75 kV to 95 kV and in the LSJ projection from 80 kV to 110 kV. The examination techniques used in some health centres differed from those of the European recommendations[3] for adult patients with regard to focal spot size (EC < 1.3 mm), total filtration (EC >3 mmAl), grid performance (EC r=8:1/40 1/cm) and nominal speed class of the image receptors (EC 400).

Radiation dose to patients

Figure 2 shows the average dose-area products (DAP), figure 3 the average entrance surface dose (ESD) and figure 4 the average bone marrow dose, gonad doses (mGy) and the effective dose (mSv) of the entire LS spine examinations for both sexes at various health centres.

Table 3 shows the average DAP, ESD, organ and effective dose for all of the 13 health centres.

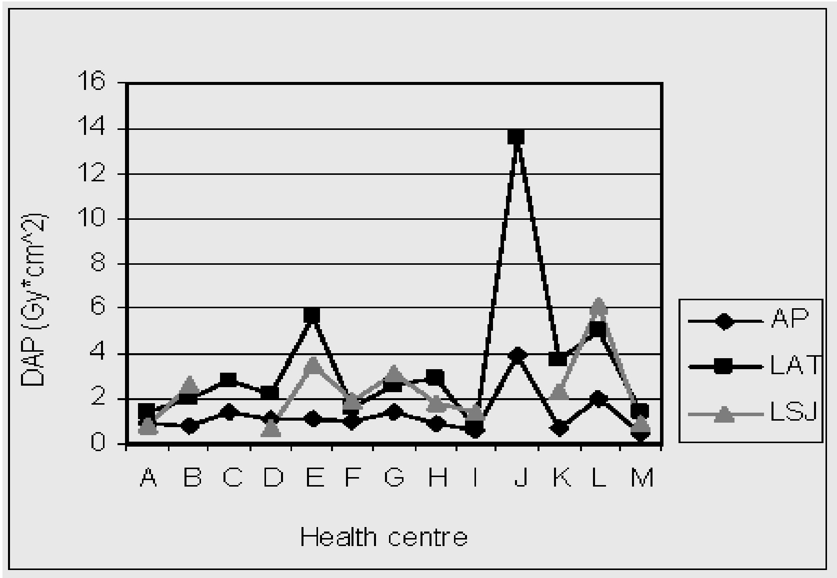

Figure 2. Average dose-area products (DAP) of the LS spine examinations at various health centres.

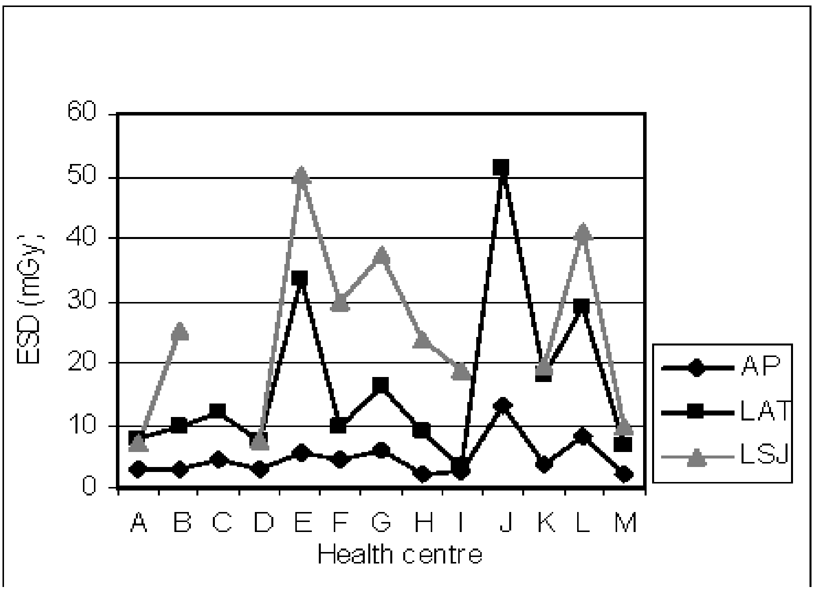

Figure 3. Average entrance surface doses (ESD) of the LS spine examinations at various health centres.

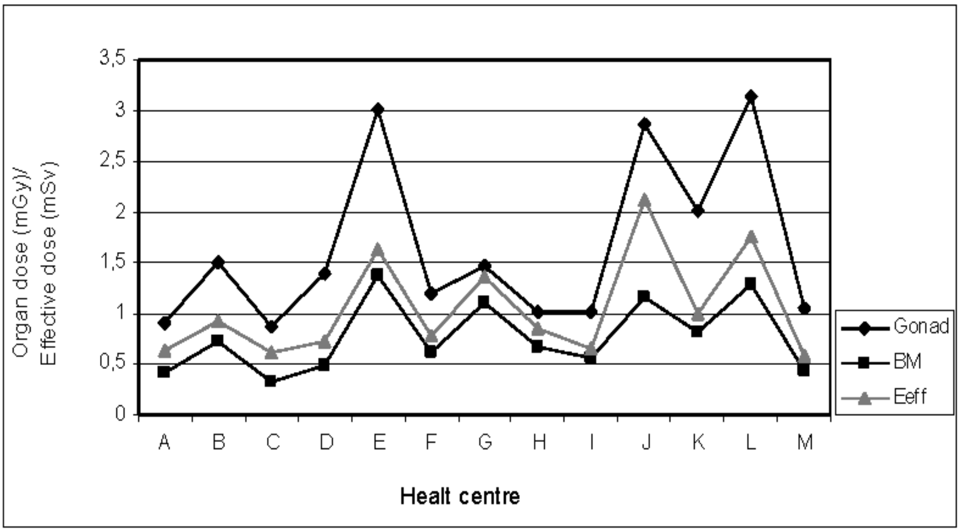

Figure 4. Average bone marrow dose(BM), gonad doses (Gonad) and effective dose (Eeff) of the entire LS spine examinations at various health centres.

Table 3. Average DAP, ESD, organ dose and effective dose for all of the health centres.

Dose	AP	LAT	LSJ
DAP (Gy*cm^2)	1.3	3.6	2.1
ESD (mGy)	4.8	17	26
Gonads (mGy)	0.4	0.6	0.8
Bone marrow (mGy)	0.06	0.36	0.43
Eeff (mSv)	0.31	0.45	0.35

The sum of the average effective dose of all projections was 1.11 mSv. The collective dose from LS spine examinations in all health centers was about 4.2 manSv. If image receptors having a speed class of 200 were replaced by image receptors with a higher speed class of 400, the collective dose would decrease by about 20%.

Conclusions

The present study on patient doses and examination techniques has shown that the doses exceed the national dose reference levels except in two health centres. The examination techniques don't meet the European quality criteria of focal spot size, total filtration, grid performance and nominal speed class of the image receptors in some health centres.

The dose variations between the various health centres were found to be high, as were the dose ranges in a single health centre. The patient dose exceeded the national reference levels in only a few health centres, whereas in many health centres the examination techniques did not meet the European recommendations with regard to focal spot size and the speed index of the image receptor. Similar results with respect to the sensitivity of image receptors were reported earlier in health centres [7]. The total dose measured in this study could be decreased by about 20% if more sensitive screens were used.

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SESSION 8. RADIATION IN INDUSTRIAL USE

Industrial applications – Accidents and regulatory instruments

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Introduction

Ionising radiation in industry is used in a wide range of applications, and it is not possible to cover all categories and fields in 20 minutes. The potential for accidents are clearly greater for radioactive sources compared to X-ray generators, and this presentation will focus on the application of sealed radiation sources.

In Norway there are 4 areas of particular importtance in this respect, partly due to the number of sources involved, and partly to the size of the sources used. These areas are:

Field of application	Typical source specification	IAEA Category	Volume and number
Irradiation facilities	40.000 – 200.000 GBq ^{137}Cs.	1	Approx. 10 smaller facilities for irradiation of blood, and one large for sterilization etc.
Industrial radiography	1500 GBq ^{192} Ir	1	Approx.150 companies and 500 radiographers, some hundreds of gamma sources.
Offshore logging	100 GBq ^{137}Cs 600 GBq ^{241}Am/Be	2	Approx. 10 companies, 50 -100 sources.
Industrial gauges	1-10 GBq ^{60}Co, ^{137}Cs, ^{241}Am ...	3	Approx. 300 companies, approx. 2000 sources

The IAEA categorization refers to IAEA Tecdoc-1191 from December 2000 (1), in which types of sources and fields of application are categorized with regard to accident risk and with regard to the regulatory requirements that should be applied.

Accident risks and scenarios

The accident risks and accident scenarios for the various fields differ.

Application	Accident risk/scenario
Irradiation facilities	Loss of shielding due to fire. Unshielded external dose rates are typical 10 Sv/h at 1 m distance. The fire department are informed of the existence of these sources. In 1982 there was one fatal accident in the large facility for sterilization, and UNSCEAR have reported 6 fatal accidents worldwide in large sterilization plants like this during a 20-year period. No accidents in the smaller Norwegian facilities have been reported to the NRPA.
Industrial radiography	Normal use includes moving the source out of and into shielded position, and accidents and incidents occur as the source may be stuck in the guide tube, detached from the control cable a.o. Accident statistics are shown later. The sources are generally under strict administrative control, with small risk of getting lost. The source and shielded container is portable, and there is thus a certain risk of theft.
Offshore logging.	Sources may get stuck downhole, and have to be abandoned – representing unintended depositing of radioactive sources beneath the seabed. Annually typically 2-5 sources are lost this way. The boreholes are cemented and closed, and there is little risk of the sources ever entering the public domain.
Industrial gauges.	Stationary sources not subjected to manipulating – thus lower risk for accidents. The administrative control by the owner may be less strict, with a potential for orphan sources due to closing of factory a.o. A typical accident involves a worker being exposed during cleaning or maintaining a tank with a level gauge in open position.

In industrial radiography accidents and incidents are frequently reported to the Norwegian radiation protection authority (NRPA). The statistics for the last 7 years are shown in the following table.

Type of accident - industrial radiography 1995 - 2002	**Number**	**%**	**Doses involved**
Concluded falsely that the exposure was completed or exposure initiated with operators working around source. -Alarm dosimeter out of use/failed-	11	28%	Large doses, up to 50 mSv
Source stuck in projection sheath	6	15%	Small doses. Less than 1 mSv
Source could not be returned to container due to mechanical failures.	7	17%	0,1-1,3 mSv
Barriers not respected	7	17 %	0 – 4 mSv
Source loose in projection sheath	5	13 %	2-6 mSv
Miscel1aneous, including 1 case of theft.	4	10 %	
Total	**40**	**100**	

The most frequent and most serious accidents happen when the source is not where it is expected to be, or when the X-ray generator is activated when it is supposed to be off.

Various types of technical failures may also cause accidents, but these are rarely serious as long as the operators know where the source is. The IAEA state that 75 % of accidents within industrial radiography are operators errors or failures to follow procedures, 25 % are equipment failures (1).

Regulatory instruments

The following traditional regulatory instruments are applicable to industrial use of radiation:

- General regulations with requirements concerning use, internal control, etc.
- Licensing and authorizations
- Requirements for appropriate levels of training
- Technical requirements to radiation sources, systems for type approvals
- Dues and taxes

Dues and taxes are frequently used in Norway in order to influence human behaviour, but are not being used in radiation protection administration. In Norway the following regulatory requirements are applied:

FIELD OF APPLICATION	IAEA category	REGULATORY INSTRUMENTS
Irradiation facilities	1	Individual licencing of each installation, with some requirements for internal control. Requirements for administration of disused sources.
Industrial radiography	1	Authorization of NDT-companies. Formal requirements for training and certification of operator (5 days training course). Requirements for internal control and procedures for handling radiological emergencies.
Offshore logging	2	Authorization of logging-company. Formal requirements for training, but no certification of operators. Requirements for internal control and procedures for handling radiological emergencies.
Industrial gauges	3	Individual authorization by licensing of each installation, may be reduced to authorization by registration. Modest requirements concerning internal control and training.

Orphan sources

The Norwegian system as described above complies with requirements and recommendations in IAEA Tecdoc 1191 (1) and the IAEA Code of Conduct (2) regarding the *safety* of radiation sources, that is the general radiation protection and emergency handling. The regulatory system has to a lesser extent been directed towards *security*, that is measures to prevent unauthorized access to, and loss, theft and unauthorized transfer of radioactive sources. In this area there is a potential for regulatory improvement, and the future regulatory system should address the security issue in a better way. Internationally, serious radiation accidents have been caused by sources entering the public domain.

Industrial gauges are particularly at risk in this context. In an ongoing project at the NRPA, an initial 20 % discrepancy was found between the number of sources registered in our database, and the number of sources actually found in the factories and facilities. After an intensive search this deficit was reduced to 10 %, and we hope to reduce it further before the project is ended. More frequent reports from the source owners to the authorities may improve the administrative control of radioactive sources.

Conclusion regarding future regulatory challenges in Norway

- The regulatory system for sealed radioactive sources should to a larger extent focus on security and orphan source hazards.
- The problem of unintended depositing of radioactive logging sources beneath the seabed must be given greater attention.

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Transportincident med iridium-192

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Summary in English

Measurements of a shipment containing Ir-192, for radiographic purposes, showed increased dose levels on arrival at the final destination in New Orleans, USA. The package containing the radioisotopes left Studsvik the 27th of December 2001 intact and in an approved transport container. Prior to departure from Studsvik, the package was measured and displayed normal dose levels.

The container was opened in February in the presence of Studsvik's personnel and different inspectors. The inspection of the container revealed that two of the three inner containers with Iridium-192 were open and the radiation was therefore able to penetrate the outer container's radiation shielding. Unclear instructions in the certificate are one of the observations. Tolerance of the inner tap is another.

An investigation of the incident is still (August) ongoing.

Inledning

Den 27 december 2001 skickades en transportbehållare innehållande strålkällor av radionukliden Ir-192 från Studsvik. Strålkällorna skulle skickas till Source Production and Equipment Company (SPEC) för tillverkning av radiograferingsutrustning.

Transporten skedde först med lastbil mellan Studsvik och Arlanda (via Norrköping). Flyg från Arlanda gick först via Charles de Gaulle Airport i Paris, Frankrike 29 december, till Memphis, Tennessy i USA och vidare till New Orleans, Louisiana, USA. Behållaren hämtades den 2 januari från flygfrakt-terminalen vid Federal Express (FedEx) lokal i New Orleans av en person anställd vid SPEC, St Rose. En dosratsmätning på behållaren tolkades först som att det var något fel på instrumentet eftersom visarutslaget går i botten. Behållaren körs därefter till SPEC:s lokaler. Vid framkomsten noteras att persondosimetern visar en dos på ca 1,6 mSv. Total dos senare bestäms till 3,4 mSv. Nya dosratsmätningar gjordes då på behållaren. Mätningarna visade dosrater på 3-4 mSv/h på ca 23 meters avstånd från locket och 0,01 mSv/h på ca 4,6 meter från den liggande behållarens sidor.

I den aktuella behållaren (54x43 cm, 126 kg) skickades 1078 st strålkällor av Ir-192

(21,5 g, totalt 366 TBq). Strålkällorna i form av små “pellets” (diameter 2,7 mm, tjocklek 0,15 mm) placerades i tre kapslar (med 375, 382 och 321 st strålkällor) av rostfritt stål som förslöts med skruvade lock. Kapslarna (med yttermått 10 mm i diameter och längden 38 mm) placerades i ett revolvermagasin (liten cylindrisk stålbehållare utan lock) och därefter i en innerbehållare av utarmat uran vilken fungerar som strålskärm. Kapslarna låg förankrade i vertikalled i laddcellen. Innerbehållaren förslöts av en plugg, även den av utarmat uran, och av ett lock med sex bultar. Innerbehållaren var placerad i en ytter-behållare av stål och stötdämpande material. Ytterbehållaren förslöts av ett lock med åtta bultar.

Den 3 januari informerades Studsvik om mätningarna av US Department of Transportation (DOT). Studsvik kontaktade direkt SSI och SKI. Den 4 januari beslutade SSI att alla transporter av radioaktiva ämnen från Isotopcentralen (IC) i Studsvik tillsvidare skulle stoppas. Detta stopp upphävs sedan först 28 februari. SSI konstaterade inledningsvis att man inte kunde utesluta att någon eller några personer fått akuta strålskador. Beslut fattades även att 7 januari genomföra inspektioner vid alla transportled (Cargo Center på Arlanda, SAS Cargo i Norrköping, samt i Studsvik). Även blodprover på viss personal togs vid detta tillfälle (dessa visade inget onormalt).

SSI gav 9 januari vissa undantag från det tillfälliga stoppet för transport av radioaktiva ämnen. Den 12 mars lämnar SSI ärendet till åklagare då man anser att Studsvik brutit mot strålskyddslagen och mot lagen om transport av farligt gods. SSI ansåg att Studsvik visat grov oaktsamhet i samband med transporten av iridium. Studsvik har idag (augusti) ingen information om tidplanen för den fortsatta processen eller resultat från den amerikanska undersökningen.

SSI häver 24 april samtliga restriktioner för Studsviks isotoptransporter.

Orsak till de förhöjda dosraterna

Behållaren öppnades i USA i närvaro av bl a Studsviks personal samt Statens strål-skydds-instituts inspektörer först den 7 februari. Vid undersökningen av behållaren framkom att två av de tre innerbehållarna med Ir-192 var öppna.

Det noterades också att behållaren inte var fastspänd på samma sätt på lastpallen som när den lämnade Studsvik. Den hade roterats ungefär ett kvarts varv och ett av de två spännbanden hade lossnat.

I de begränsningar och villkor som angivits i typgodkännandet av transportbehållaren (2835A) stod bl a:

- att kollit är godkänt för ett innehåll av Ir 192 i form av "pellets" i fast metall.

- att kollit är godkänt för ett högsta aktivitetsinnehåll av 370 TBq. Innehållet vid händelsen var 366,04 TBq.

- att kollit är godkänt för en högsta tillåten effekt av 57 W från strålningen.

- vidare angavs i godkännandet att pellets antingen skulle inneslutas i kapslar med skruvat lock (screw-topped) eller i slutna (sealed) kapslar. Dessa skall vara konstruerade så att de enbart kan öppnas genom att kapseln förstörs. Det konstaterades att formuleringen i det engelska godkännandet inte var entydigt: "The pellets shall be contained in either screw-topped capsules or sealed capsules constructed so that they can only be opened by destroying the capsule.". Från 18 januari har DOT lagt till texten "screwtopped capsules with sealed welded caps are acceptable". Efter ändringar av ägaren till behållaren, CROFT, har engelska transportmyndigheter den 1 juli godkänt förändringen "…either sealed metal capsules constructed so that they can only be opened by destroying the capsule, or in screw-topped metal containers wich have been lock welded or otherwise positively fastened to ensure that the closure cannot become loose or detached under routine, normal or accidental conditions of transport".

- i godkännandet angavs att lämpligt förpackningsmaterial (suitable packing) skall användas för att förhindra större rörelser (gross movements) av kapslarna samt inbördes stötar. Förpacknings-material som förhindrar vertikala rörelser har använts. Horisontellt kunde kapslarna dock röra sig mindre än 1 cm. Även denna text har CROFT nu ändrat till "Suitable packing material must be fitted within the cavity to prevent radial or axial movement of the primary containers within the shielding pot during transport".

Studsvik hade under hösten 2001 konstaterat att en liknande observation inträffat vid en tidigare transport, men då hade inte det skruvade locket helt gängats upp.

Förbättringar i kvalitetssystemet har gjorts bl a genom tydligare instruktioner.

Iridium

Iridium är en silvervit, spröd, extremt hård metall. Den har atomnummer 77 och dess kemiska beteckning är Ir. Atomvikten är 192,22 och densiteten 22,6 kg/dm^3 vilket gör att Ir jämte osmium (Os) är världens tyngsta grundämne. Ir smälter vid 2410 °C och kokar vid 4130 °C. Ir förekommer ofta naturligt i form av legeringar med platina (Pt) och Os för att göra metallen hårdare.

Den vanligast förekommande radioaktiva isotopen av Ir är Ir-192 som sönderfaller till Pt-192. Den framställs genom att naturligt Ir (Ir-191, 37,3 % och Ir-193, 62,7 %) bestrålas med neutroner (n-gamma-reaktion). Även Ir-194 bildas men har mindre tvärsnitt och kortare halveringstid. Ir-192 har halveringstiden 74 dygn vilket gör den lämplig att användas som strålkälla vid material-undersökningar och olika patientbehandlingar.

Mätresultat

TL-dosimetrar för de två personer I Studsvik som utförde arbetet med att färdigställa transportbehållaren utvärderades direkt. Djupa persondos-ekvivalenten, Hp(10), för de två personerna var 0,0 mSv respektive 0,2 mSv under december månad. Inget av de registrerade värdena avviker från förväntade värden.

Vid arbetet i IC-lokaler har inga lokala strållarm eller installerade detektorer indikerat förhöjda strålnivåer. Fast placerade strållarm finns t.ex. i de lokaler där transportkollit färdigställdes.

Enligt transportdokumentet skall högsta dosrat vid ytan respektive på 1 meters avstånd från transportbehållaren anges. Mätningarna utfördes med ett instrument av typ Automess AD5. Dosraten vid ytan uppmättes till 0,6 mSv/h medan den maximala dosraten på 1 meters avstånd uppmättes till 0,02 mSv. Mätningarna utfördes enligt mätinstruktion och görs på alla sidor av kollit. Således har ingen indikation på onormala dosrater erhållits. Ingen onormal ytkontamination uppmättes ($\beta/\gamma < 4$ Bq/cm^2, $\alpha < 0{,}4$ Bq/cm^2).

Fast i rummet monterade TL-dosimetrar utvärderades också. De avlästa värdena på miljödosekvivalenten var 0,6 mSv respektive 1,8 mSv. De avlästa värdena för december månad var ett av de lägre värdena under 2001 och indikerade inte några onormala strålnivåer.

I närheten av de aktuella arbetsutrymmena finns känsliga utpasserings-monitorer innehållande gasdetektorer med stora detektorytor som larmar om kraftigt förhöjda dosrater förekommer i lokalerna. Dessa har inte larmat under den aktuella månaden.

Två utomhus placerade dosimetrar vid utfarten från Studsvik visade heller inget onormalt utan överensstämmer med vad som kan förväntas vara normala bakgrundsvärden.

9/1 meddelar FedEx France att en doskontroll av kabinpersonal i Paris inte visat något onormalt.

Inte heller i Sverige konstateras några stråldoser.

Strålskyddsmätningar vid SPEC benämnd “SPEC Event Summary”, redovisade några mycket höga dosrater från locket. Även i sidled registrerades strålning från emballaget. Mätresultaten redovisas som citat från rapporten. I rapporten redovisas också mera detaljerade resultat av persondosutvärderingar i USA. Slutsatsen av dessa, förutom de 3,4 mSv till chauffören, blev att ingen person utsatts för höga persondoser i USA.

Franska myndigheter har låtit utföra blodundersökningar på tranportpersonal i Paris. Resultat av dessa prover anses visa att två personer fått stråldoser (15 och 100 mSv) med stora osäkerhetsintervall. Ytterligare information och förutsättningar för dessa mätningar är inte kända.

Occupational exposures – Radiation Protection at Swedish Nuclear Power Plants. The Authority's retrospective view.

Ansi Gerhardsson and Lars Malmqvist
Swedish Radiation Protection Authority

Occupational dose reduction at nuclear power plants (NPP's) is important. The importance comes not only from the health and safety of the labour force. The associated requirement for a good management system enhances the safety, quality and reliability of the installation and thus the finance of the plant.

The Alara concept (As Low As Reasonable Achievable) includes social and economical factors. To strengthen the connection between occupational exposures and economy the nuclear power industry has put a price tag on occupational doses. The utility value of 1 averted manSv has the price of 4 million SEK (i.e. about 440 k EUR).

Radiation protection and reactor safety is closely linked. Good radiation protection coupled with good safety practice enhances the positive effects in both areas. A well-planned job results in a job performed accurately with good speed, high quality and security. The need for re-work is minimized and the occupational exposures can be kept low. Radiation protection should always be a natural ingredient in every project at a nuclear power plant.

Comparisons and events

In Swedish nuclear power history occupational doses have been kept low. Comparisons with other countries show good values until 1992. While the doses in France, Germany, Japan and the USA, which started high, began to decrease steadily, the traditionally low doses at Swedish plants began to rise. One reason for the low doses during the second half of the 1980's was the less amount of maintenance in combination with good planning performed at the yearly outages.

Between 1992 and 1997 an increasing dose trend could be seen at Swedish NPP's. Two factors are noted. More maintenance was needed due to the aging of the oldest reactors. The Swedish Nuclear Inspectorate (SKI) required more extended inspection programs. The first generation of Swedish boiling water reactors (BWR) Oskarshamn 1 and Ringhals 1 were mostly affected.

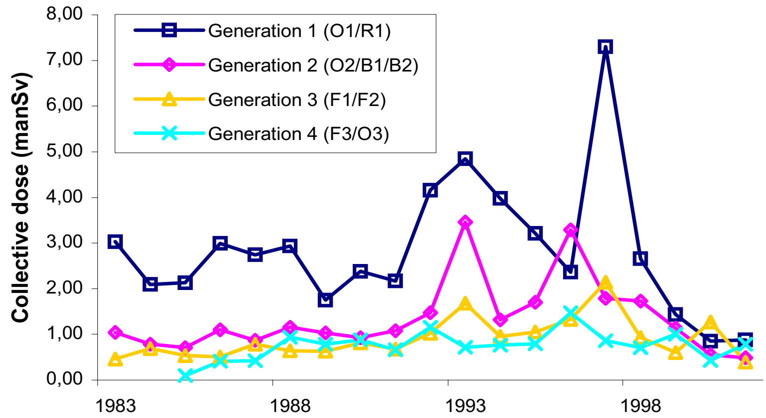

Figure1. Collective dose for different generations of reactors.

In 1992 the so-called strainer-event occurred, when insulation material was ripped off causing clogging of the strainers in the safety injection systems. This event led to considerable modification of all the BWR's with similar design. All fibre insulation in the reactor containment was replaced by metallic insulation. New safety analysis reports were produced, plans for modernisation of the reactors were created and new test methods were invented. These efforts contributed to the higher occupational exposures at BWR's during 1993 and 1994.

In 1995 a change of steam generators at Ringhals 3 (a pressurised water reactor, PWR) was performed and in 1997 the SPRINT-project (Säkra

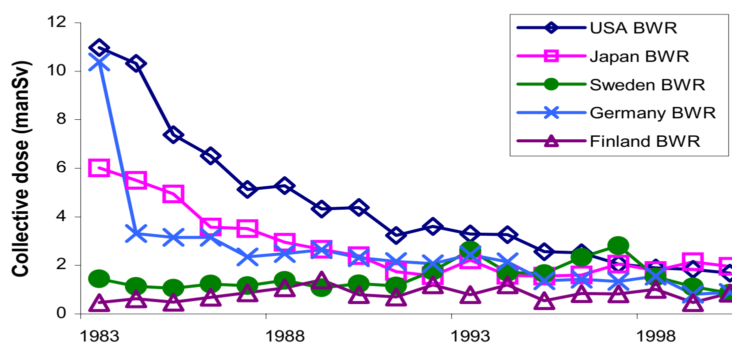

Figure 2. Annual collective dose per BWR.

PRimärsystemets INTegritet) at Ringhals 1 was carried out. The SPRINT-project aimed at securing the integrity of the primary system below the core to make it comparable with the safety level at reactors with internal pumps. Lower doses were obtained because of fewer measures due to problems with inter granular stress corrosion cracking, less amount of testing and reduction of stellite.

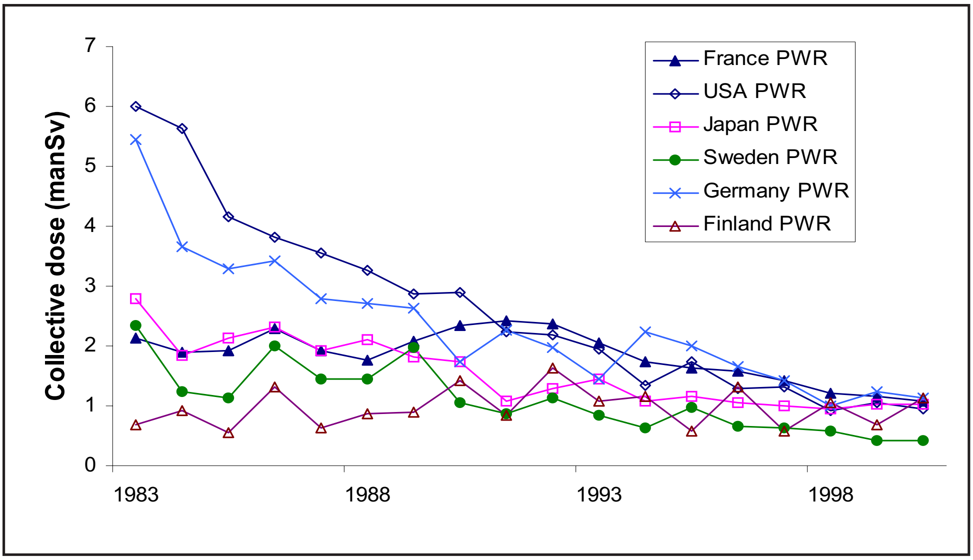

Figure 3. Annual collective dose per PWR

The dose trend at Swedish NPP's changed in 1997 and since then the collective doses have steadily been decreasing. The radiation environment at the nuclear power plants has changed. For several of the reactors decreasing dose rates have been seen as a result of change in reactor water chemistry and the overall picture shows better circumstances regarding dose rates at all the plants.

Measures taken by the authority

The Swedish membership in the European Union brought about changes in the efforts made by the Swedish Radiation Protection Authority (SSI). The SSI collection of regulations had to be rewritten to include the directives from 96/29/Euratom. These regulations include for example dose limitation, medical examinations, effluents, waste and filing of documents.

The dose limits for occupational exposures were kept at 50 mSv per year with a maximum of 100 mSv per five-year period. At that time SSI choose to exclude the former limit of 700 mSv for a lifetime. The five-year doses have

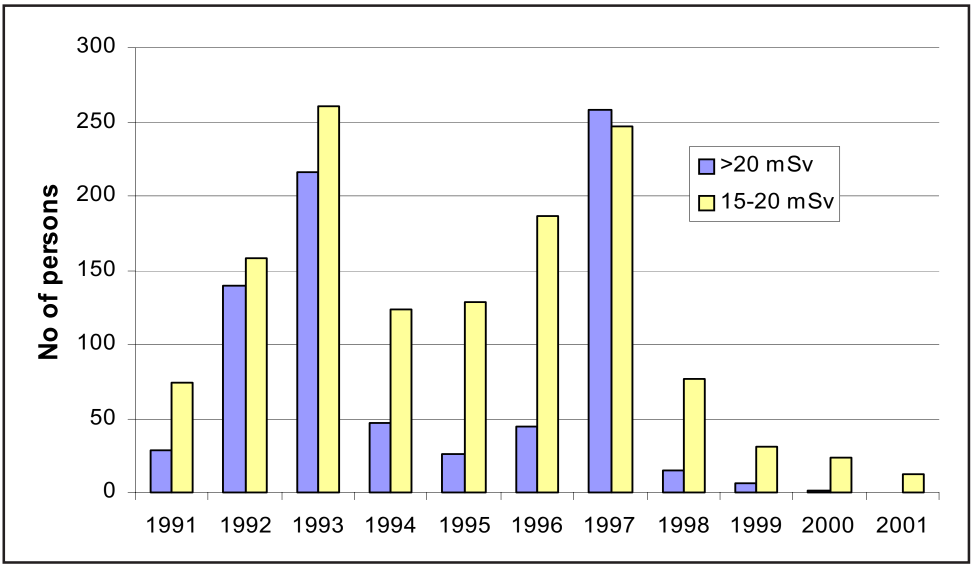

Figure 4. No of persons in higher dose intervals.

been descending during the years and last year, 2001, no one received a five-year dose exceeding 70 mSv.

SSI states in the regulations that the work shall be performed in such a way that human exposures are limited as far as reasonably achievable with social and economical factors taken into account. For this purpose the licence holder shall ensure that goals and needed actions for control are established and documented and that needed resources are available. The goals and actions of control shall be adjusted with respect to the prerequisite of the plant and be drawn up to take care of the daily as well as the long-term radiation protection. The practice, including the goals and actions of control, shall regularly be followed up and evaluated and documentation on the evaluation shall be sent to SSI.

The licence holders are expected to develop and implement Alara-programs. The plant management shall strongly declare the importance of radiation protection all through the organisation from top to bottom and secure that the labour force knows and abides by the radiation protection policy. SSI requires education and training programs for all persons working with matters related to radiation protection, operation and maintenance. This includes external workers holding posts as foremen. The education shall be adjusted with respect to the kind and extent of the work as well as the environment where it is performed.

The licence holder shall appoint a radiation protection manager. The manager shall hold sufficient competence on matters related to radiation protection and shall be the qualified expert on radiation protection at the plant. She, or he, shall actively work for the complying of regulations and conditions. The manager shall see to that the work at the plant is performed at an acceptable level for radiation protection. She or he shall also ensure that local rules are established in order to avoid unacceptable or unnecessary doses.

To have preparedness for the unwanted but possible problem such as fuel damage is an important task in the nuclear power plant radiation protection. Fuel damage can cause increased dose rates and result in problems at outages. SSI prescribes that the licence holders shall have a documented policy in case of fuel damage, including the plant's strategy for avoiding fuel damages to the most reasonable extent as well as a strategy to handle the situation would a fuel damage occur.

During the 1990's SSI have financed research projects in the areas of effects of reactor water chemistry and different methods for reduction of dose rate by decreasing activity build-up on surfaces and fuel decontamination.

Both the authority and the licence holders benefits from an open and constructive dialogue. The Radiation Protection Authority can contribute to the exchange of experiences between nuclear power plants and authorities within and outside Sweden. SSI participates in the international radiation protection community through IAEA, OECD/NEA and the EC/European Alara Network.

Measures taken by the industry

The nuclear power industry has, apart from the requirements from the SSI, implemented several measures intended to decrease occupational exposures.

Until a few years ago dose rate measurements were performed only during the yearly outages. Currently measurements are performed continuously and changes in dose rates can be directly connected to events and better analysis of operating conditions, like water chemistry, can be achieved.

Many efforts have been made to improve the reactor water chemistry and to increase the water quality. Cleanliness while working with open system together with high water quality decreases the risk of activating particles and the risk for fuel damages.

The nuclide contributing mostly to occupational doses is ^{60}Co. The largest source of cobalt is stellite. Stellite is an alloy frequently used in valves. Decreasing the amount of stellite used will reduce the amount of cobalt released to the reactor water and thus lower the radiation levels.

The future

The nuclear power industry's plans for the future makes it clear that we can expect an increasing extent of maintenance and modernisation in the next few years to come. This will probably result in higher occupational exposures. Safety findings and related events at NPP's within and outside Sweden can lead to an increasing amount of testing and maintenance at the Swedish nuclear power plants.

The Radiation Protection Authority expects the open climate for dialogue and development and research cooperation in radiation protection issues between the authority and the industry to continue. The ambition to optimise the radiation protection at the plants is in the interest of both parties. It is the task of the authority to make sure the licence holders follows stated laws and regulations to obtain this goal.

Posters

8a A brief description of the OMINEX project "optimisation of monitoring for internal exposure"

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Summary

OMINEX (Optimisation of Monitoring for Internal Exposure) is an on-going international project part-funded by the European Commission within the 5th (EURATOM) Framework Programme.

Monitoring of the workforce in the nuclear industries is carried out primarily in order to demonstrate compliance with European Union Basic Safety Standards for the protection of the health of workers against the dangers arising from ionising radiation. There is however no compilation of information on internal dose monitoring programmes currently in use in the EU countries.

The aim of **OMINEX** (**O**ptimisation of **M**onitoring for **In**ternal **Ex**posure) is to provide advice and guidance on designing and implementing internal dose monitoring programmes in the workplace, in such a way that best use is made of available resources, while minimising costs. Advice will be provided on both routine and incident (special) monitoring for a range of specific examples of exposures to industrial materials. Topics to be covered will include overall design of monitoring programmes, choice of monitoring method(s), (e.g. excretion

monitoring vs. in vivo monitoring), choice of measurement technique (e.g. alpha spectrometry vs. mass spectrometry), monitoring intervals, measurement frequency, required measurement sensitivity, and resulting uncertainty in assessed intakes and doses.

The project is being carried out by a consortium of European research/ advisory organisations working together with nuclear industry organisations: in France, these are the Institut de Radioprotection et de Sûreté Nucléaire (IRSN), Electricité de France (EdF) and Commissariat à l'Energie Atomique (CEA); in the UK, the National Radiological Protection Board (NRPB), with additional input from the UK nuclear industry; in Finland, the Radiation and Nuclear Safety Authority (STUK) and Teollisuuden Voima Oy Nuclear Services (TVONS); and the Belgian Nuclear Research Centre (SCK·CEN). The project has been underway for 20 months, and will conclude at the end of 2003, at which time a training course will be held to disseminate the results of the project to radiation protection professionals with responsibilities in this field.

Scientific and technological objectives

The central objective of OMINEX is to provide advice and guidance on the design and implementation of internal radiation dose monitoring programmes in the workplace, so that best use can be made of available resources, while minimising costs. Advice will be provided on both routine and incident (special) monitoring for a range of specific examples of exposures to industrial materials. Topics to be covered will include overall design of monitoring programmes, choice of monitoring method(s), (eg excretion monitoring vs. *in vivo* monitoring), choice of measurement technique (eg alpha spectrometry vs. mass spectrometry), monitoring intervals, measurement frequency, required measurement sensitivity, and resulting uncertainty in assessed intakes and doses.

The broad approach to optimisation of monitoring programme design will be an evaluation of costs versus "benefits". The latter will be quantified primarily by assessing the accuracy with which intakes and doses are determined. Current practice in internal dose monitoring, and the associated costs of monitoring, will be investigated, as well as the accuracy of assessed intakes/doses . Work on development of advice and guidance is also conducted.

The main objectives are:

• to collect information on, and provide a critical evaluation of, arrangements for internal dose monitoring in nuclear and non-nuclear industries
• to collect and evaluate information on the costs of internal dose monitoring

• to compile descriptions of the bioassay and *in vivo* measurement procedures used by EU laboratories for monitoring of workers from the nuclear industries, and investigate how the available methods and techniques could be exploited in such a way as to reduce the uncertainties in measurements
• to investigate and quantify the major sources of uncertainty in internal dose assessed from the results of particular monitoring methods and measurement techniques
• to provide advice and guidance on the design and implementation of monitoring programmes

Scientific and technical assessment

Survey of internal dose monitoring programmes

The questionnaire for the survey of internal dose monitoring programmes requests information on direct (*in vivo*) and indirect (bioassay and air sampling) monitoring for internal exposure to fission and activation products, and to compounds of uranium, thorium, plutonium, americium, mixed oxide (MOX) fuel, and other actinides. Separate sections cover routine and incident (special) monitoring. The complete questionnaire is quite long (55 pages in total). To encourage Dosimetry Services to respond, only those parts of the questionnaire that are relevant to the work of a particular organisation have been sent out. A one-page “pre-questionnaire” was used to establish the radionuclides monitored and monitoring methods used by each organisation. Thus, a nuclear power station operator carrying out *in vivo* and bioassay monitoring for fission and activation products would receive a questionnaire of only 14 pages.

It was decided that the questionnaire should be sent to every country within the European Union (EU) where internal dose monitoring is carried out, and that coverage of the “Associated States” of the EU should be as great as possible.

A database for storage and reporting of all information gained was designed using Microsoft Access 97™. The database contains 39 tables arranged according to the subtitles in the questionnaire, starting with general aspects, continuing with data on monitoring methods for fission and activation products and actinides, and ending with data on dose assessment methods and dose statistics. All results from the survey will be kept anonymous.

Data based on information provided both in the pre-questionnaire and in the full questionnaire show that, at nuclear power plants, fission and activation products are monitored with whole-body counting. Some organisations also monitor tritium in urine routinely but only three with a weekly frequency. All organisations report that they have well-defined pre-planned arrangements for internal dose monitoring. Some also report that they sub-contract certain monitoring to outside organisations. In most organisations monitoring is carried out to assess individual doses for entry onto a legal dose record. In many organisations monitoring is also carried out to reassure the workers that they are not receiving excessive doses.

When applicable, whole-body and thyroid counting is used also for incident monitoring, complemented with other methods.

Dose calculations are done primarily manually or using own software. Of commercial dose calculation programs, most popular is LUDEP followed by IMBA.

Costs of internal dose monitoring programmes

The questionnaire on costs of internal dose monitoring requests information relating to the costs of individual and collective monitoring (measurement procedure used, the price of each measurement, number of analyses performed for each worker, capital cost of equipment used). Information has been requested mainly on costs of monitoring for exposure to actinides. All results from the survey will be kept anonymous.

Uncertainties in measurements

Assessment of internal dose relies on results obtained from direct measurements such as whole-body, lung or thyroid monitoring and from indirect measurements such as bioassay of urine and faeces For the survey of measurement uncertainties in bioassay monitoring, the targeted analytical procedures were those set up for monitoring actinides, while the *in vivo* monitoring survey additionally covered fission and activation product monitoring. For bioassay and *in vivo* monitoring the answers gathered were provided mainly by laboratories that make these measurements routinely.

Uncertainties in internal dose assessments

The approach adopted in OMINEX is essentially to optimise the design of a monitoring programme by minimising uncertainties in assessed intakes and doses (subject to a consideration of costs and other factors). This requires the development of a methodology for assessing confidence intervals on assessed intakes and doses that takes account of the major sources of uncertainty. Factors contributing to the total uncertainty can be classified under three headings:

- uncertainties in measured bioassay quantities
- uncertainties in model parameter values describing the biokinetic and dosimetric behaviour of the radionuclide(s)
- uncertainties in the assumed intake pattern (time of intake(s), acute vs. chronic, etc.).

Methodology for the determination of confidence intervals on assessed dose

Determination of total uncertainties arising from the use of each of the available monitoring methods for a particular radionuclide and/or compound will enable specific issues to be investigated, including :

General

- What is the best choice of monitoring method for a particular radionuclide / compound?
- Is there a benefit to using more than one monitoring method?
- How does total uncertainty vary with the number of measurements?
- Incident monitoring
- For an acute intake, should measurements be performed at fixed time intervals, or should measurements be more frequent soon after the intake?
- If the time of intake is unknown, what effect does this have on the optimum number and times of measurements?

Routine monitoring

- How does total uncertainty vary with monitoring interval?
- What is the best assumption to make about the time(s) of intake (eg single intake at mid-point; continuous, uniform chronic)?
- If a single intake in each monitoring interval is assumed, should a time other than the mid-point be assumed?

Conclusions and acknowledgements

Work in the OMINEX project is continuing and the consortium considers that all the stated objectives will be met. Answers to the questionnaires were not very easy to get although we tried to approach people known to us earlier. Enough information to meet the requirements is however available. We thank all our colleagues for giving us the information we asked for. Without their help all our conclusions would have been based on published information only.

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8b Radioactive sources in Sweden: a comparison between registered data and reality

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Background and method

In connection with the restructuring of the licensing procedure for Industrial gauges in Sweden a questionnaire was sent out to 1 100 licence-holders asking them to report on differences between the registered information they had received and the physical reality.

The total number of licences of this kind is about 2 500 in Sweden. The new procedure of licencing is intended to reduce the work at the authority by a simpler registration procedure. This will be done by giving only one licence to a holder of sources and to register the information on sources by a separate registration procedure. Previously, the holder of sources (licence-holder) was in most cases given a licence for each source.

Result

The result of the questionnaire can be presented in the form of the following graph. Most answers indicate discrepancies between the registered information and reality. New contact persons were given in 78 % of the answers, 20 % were corrections of source-related information, 14 % were new addresses, and 8 % were difficult to interpret. Finally, reorganization of the holder-company was indicated in 15 % of the answers. For those answers that could easily be interpreted (about ¼) the sum of the registered source-activity was about 1 TBq too low.

Conclusion

From a supervisory point of view the discovered discrepancies were alarming. The new way of licensing will now include an annual confirmation from the licence-holder of the accurateness of the information registered at the authority.

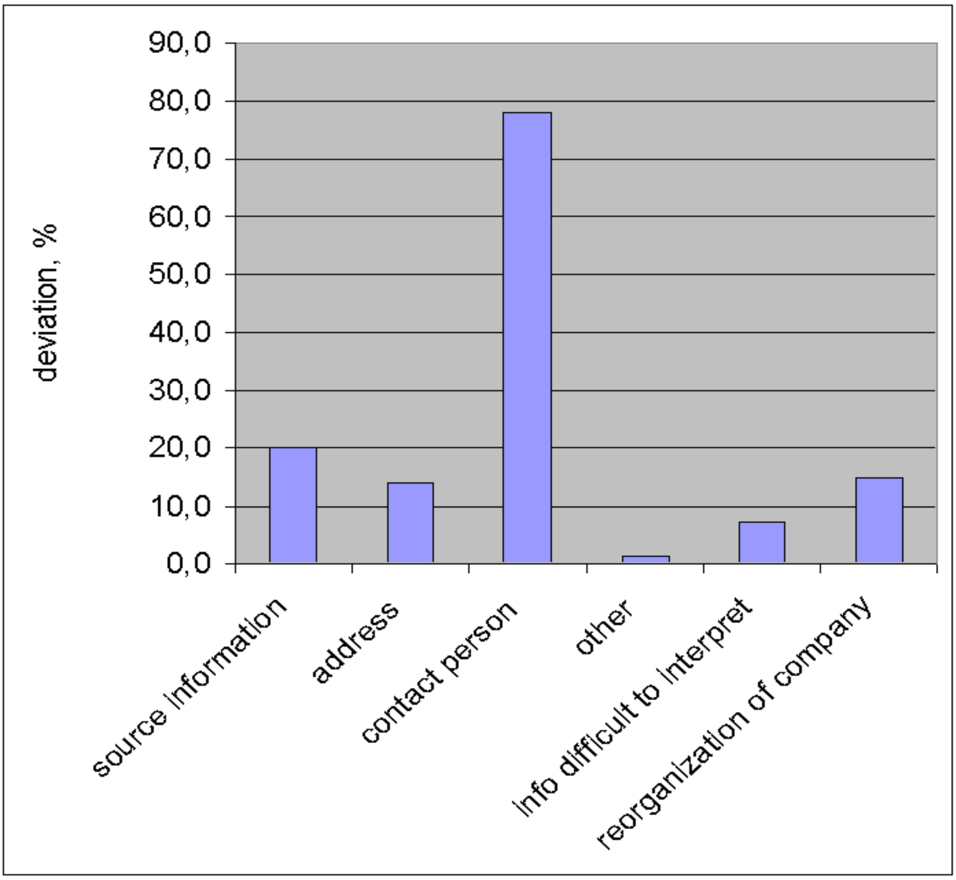
90,0
80,0
70,0
60,0
50,0
40,0
30,0
20,0
10,0
0,0
deviation, %
source information
address
contact person
other
info difficult to interpret
reorganization of company

Figure 1.

SESSION 9. TRAINING AND CERTIFICATION

Practical experiences in educating radiation safety officers

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Introduction

The Finnish Radiation Act[1] requires that a party applying for licence to run a radiation practice shall submit a description of organization, *inter alia* naming the radiation safety officer and describing the qualifications of other persons involved in the use of radiation. The Radiation and Nuclear Safety Authority (STUK) defines the level of qualification required for radiation safety officers and other persons participating in radiation work or authorized to use radiation sources independently. The fields of competence are[2]:

- Medical use of radiation;
- Use of radiation in industry, research and education; and
- Sale and maintenance of radiation sources.

Each field is subdivided into several subfields or levels of competence. Prior to 1992 – when the Act became effective – one could only qualify as radiation safety officer before a special examination panel, but now it is possible for STUK to delegate this responsibility to certain educational institutes. At present 44 such units have this right, 25 of them in the medical field[3].

The course in radiation protection

The Laboratory of Radiochemistry, University of Helsinki, has since 1992 offered a course in radiation protection. The course comprises ten hours of lectures, a written examination and two laboratory exercises, and its aim is the competence as radiation safety officer for using open and sealed radiation sources in industry, research and education. This course has been quite popular, 742 persons have participated, of whom 542 have received the certificate of competence (1991—2001). Most participants have been university students or university personnel, but participation is not restricted to these groups. Lectures have been offered once or twice annually, and the course has also twice been held for postgraduate students of the Medical Faculty.

Textbooks

There is no lack of textbooks on radiation protection, but most of them still use the old units rad, rem and curie. Even Marttila's excellent books in Finnish[4, 5] were written before the EU Directive on Basic Safety Standards[6] (BSS), and are thus partly obsolete. It would not be realistic to attempt to teach students the abstract concepts of dosimetry if they are further complicated by different systems of units and quantities, and terminology like 'dose equivalent' against 'equivalent dose'. It was necessary to compile a textbook or compendium[7] for the course, and also to update it continuously. The contents of the compendium are:

- Definitions, quantities and units, the detrimental effects of radiation, laws and directives, safety licences, dose limits, calculating doses, practical aspects of radiation protection, radiation protection measurements and handling radioactive wastes;
- The Radiation Act and Decree;
- Ranges of particles;
- Absorption and build-up factors;
- The radiation characteristics of some 100 commonly used radionuclides;
- The absorption classification, dose conversion factors, exemption activities, ALI_{min} and MAC values for the same radionuclides;
- An example of a radiation safety data sheet for a radionuclide, and
- Contact data of the Authority (STUK).

The written examination

The examination format differs from the usual examinations at the university. It is not intended to test the memory of the examinee, but rather his or her ability to apply rules and calculations to realistic situations in dealing with radiation sources. In such situations the examinee has the role of radiation safety officer. The examinees can thus use the compendium during the examination. Usually five tasks are set, e.g.:

- Calculate the external or internal dose from given parameters. Suggest appropriate action if an employee has received the dose calculated.
- Compile safety instructions for a task involving a certain radiation

source. Alternatively, find the errors in a researcher's plan for doing such a task.

- Find the errors in a series of pictures depicting handling of an open source.
- An accident or incident has occurred in the laboratory, which may have exposed personnel to radiation or may have contaminated the premises. What should the safety officer do? Examples of accidents are found in the literature, or the examiner can draw on his or her experience.
- Explain how radioactive waste of given composition should be rendered innocuous.

This type of examination gives a reliable indication of the examinee's ability to act as a safety officer. The questions may be arduous to set up, but judging the answers is easier than judging the more usual essay-type exam. On an average 60% of the participants in each examination pass.

Laboratory exercises

There are two exercises, together taking roughly one working day. Radiation protection instruments and simple dosemeters are demonstrated, and the participants calibrate the meters, and scan surfaces for simulated contamination. Safe handling of open and closed sources is demonstrated and practised. The participants write reports on the exercises.

Practical experiences

The course 'Radiation protection' originated as in-house safety training for students of radio-chemistry. Under the new system of training safety officers introduced under the Act of 1991 the course was aimed at providing formal competence, which at first was a prerequisite for taking the subsequent laboratory courses in radiochemistry. This system ensured that all students had sufficient knowledge of safety procedures before handling radiation sources in the laboratory. On the other hand many students experienced great difficulties in passing the stiff examination for safety officer, since many of them never had seen a radiation source or a radiation laboratory. This problem was solved by restructuring the curriculum, so that a student starting to specialise in radiochemistry takes the following courses after passing the intermediate level in Chemistry:

1. Basic Radiochemistry I, with written examination
2. Lectures in Radiation Safety

3. Short exam in Radiation Safety, of the multichoice quiz type
4. Laboratory exercises in Radiation Safety, with written reports
5. After passing the short exam and completing the exercises the student receives one credit unit for Radiation Safety and can participate in laboratory exercises with radiation sources.
6. It is recommended that students taking Radiochemistry as their main subject take the examination for Radiation Safety officer after participating in the laboratory-oriented courses Basic Radiochemistry II, Nuclear Spectrometry and Analytical Chemistry of Radionuclides. In any case students are required to attain competence as Radiation Safety officers before they may work independently in the laboratory, e.g. as summer trainees or doing their Master's theses.

This curriculum naturally only applies to undergraduate students. Other participants attend the course with the sole aim of attaining formal competence. They usually have practical experience of handling radiation sources and therefore experience less difficulties in passing the examination for safety officer.

The examination for Radiation Safety Officer should judge the examinee's ability to *apply* rules and formulae of radiation protection to realistic situations, rather than his or her ability to *remember* concepts learnt by heart.

Feedback from the Authority

The course curriculum was originally planned in cooperation with experts from STUK. The Authority in 1996 and 2001 requested institutions licensed to issue certificates of competence to report in detail on courses and examinations held: numbers of participants in examinations, percentage passed, questions used in examinations and numbers of certificates issued. Unfortunately STUK has not had resources available to use the reports as feedback to the institutions. Digests of the best examination questions would have been especially welcome.

Discussion

Radiation protection and work with radiation require professionals at three levels of expertise:

- Persons who may perform specified tasks with radiation sources without supervision, but do not supervise the work of others (responsible users);
- Radiation safety officers, appointed by the licensee and responsible for the safe use of radiation; and

- Qualified experts, as defined in the BSS directive[6].

It is the duty of the safety officer to ensure that the technical personnel referred to in the first category receives adequate training. There are not, however, enough training courses available for non-academic personnel.

There is an adequate selection of courses and examinations for presumptive safety officers. The third category, qualified experts, is not yet covered by ST-guide 1.4[2], but will be considered in a future revision. For the medical use of radiation the hospital radiation physicists are considered to have qualified expert status.

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Education, Training, and Qualification of Radiation Users in Finland

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Responsibilities of licence holders

Radiation Act (592/2001) stipulates responsibilities of licence holders to ensure education and training required for the radiation users. Workers shall be provided with training and instructions for their duties appropriate to the kind of work and to conditions at the workplace. Particular attention shall be paid to information on the health hazards of radiation and on safety-enhancing work procedures in order to prevent unnecessary exposure to radiation and events leading to exceptional exposure thereto.

Workers shall comply with the instructions issued, and shall also otherwise attend to their own radiation safety and to that of others.

An application for a safety licence shall be supplemented with a description of the radiation user's organisation (an organisation description), specifying the name of the radiation safety officer responsible for safety in the use of radiation.

Having regard to the nature and scope of use of the radiation and to conditions at the place of use, the organisation description shall also provide sufficient information concerning:

1) the competence of the persons participating in the use of the radiation,
2) the duties and division of responsibility pertinent to the safe use of the radiation,
3) other arrangements to ensure safety at the place of use of the radiation.

The Radiation and Nuclear Safety Authority - STUK shall stipulate the qualifications required of a radiation safety officer and of other persons working in the user's organisation, and shall investigate compliance with the said stipulations. STUK has issued the guide ST 1.4, "Organization for the use of radiation", which is under revision. The Guide ST 1.7 ”Radiation safety training in health care", is under preparation. A guide concerning the organizations giving radiation safety training will be prepared in the near future.

Qualifications in Health Care

The Requirements of Directive 97/43/Euratom concerning medical exposures have been implemented to the Finnish radiation legislation by the Amendment (1142/1998) to the Radiation Act (592/1991) and by the Decree of the Ministry of Social Affairs and Health (423/2000). Qualifications of the workers involved in the use of radiation in health care have been stipulated in that Decree as following:

Responsibility for measures

A physician responsible for a measure causing exposure to radiation shall possess qualifications consistent with the character of the measure for assessing the justification and optimisation thereof, and also contributing to the interpretation of the results of the measure. The practitioner shall be responsible for meeting the qualification requirements, and the general qualification requirements to be considered when assessing these shall be:

Radiotherapy: a specialist in oncology or other specialist qualified for radiotherapy

Isotope medicine: a specialist in clinical physiology and isotope medicine or other specialist qualified in isotope medicine

X-ray examinations and surgical radiology: a specialist in radiology. Other consultant physicians responsible for the use of X-ray equipment shall possess the knowledge of radiation protection necessary for measures performed in their specialisms. When radiation protection has formed no part of the studies of a physician the required knowledge may be demonstrated through the separate radiation protection training, which fills the requirements defined by STUK.

Dental X-ray examinations: a dental surgeon or other physician with basic knowledge of the health effects of ionising radiation and of the exposure of the patient to radiation.

A physician other than those referred to in the foregoing who is responsible for measures shall possess basic knowledge of the health effects of ionising radiation and of the exposure of the patient to radiation in the measure concerned.

Performance of measures causing exposure to radiation

The qualification requirements for a physician responsible for measures shall apply when a physician performs a measure causing exposure to radiation.

An X-ray assistant (radiographer) may independently perform an X-ray imaging according to a referral. Under the supervision of a physician responsible for measures other professional health care staff may assist in the use of X-ray equipment for the use of which they have been properly trained.

A person who has vocational education in dental X-ray photography may perform dental X-ray imaging according to the instructions of a physician.

Any other person participating in the performance of measures causing exposure to radiation shall possess training and experience in accordance with the nature of the said assignment.

Radiation safety training in vocational education

Radiation safety training in vocational education is the duty of universities, vocational high schools and other training organizations. The licence holders are responsible for the complementary training of workers involved in the use of radiation. The ST guide 1.7 concerning the radiation training in health care will give STUK´s opinion on the content and minimum hourly amounts of the radiation safety training in health care. STUK has asked outside comments to the draft of the guide from the training professionals and other contact groups. We got lot of comments and improvements to the guide. The main opinion was that the guide is needed and the topics which were proposed in the draft to include in the radiation safety training was considered appropriate. But changes to the hourly amounts were proposed in the comments.

Radiation safety training of radiation safety officers and operators in charge

STUK has accepted some training organizations to give certificates of radiation safety officers. The certificate is an expression that a person has past an exam of the radiation safety officers and he or she can be appointed as radiation safety officer of a radiation practice. When granting a radiation safety licence for the use of radiation STUK accepts the radiation safety officer nominated by the responsible party.

The exam of radiation safety officer is nowadays included in the specialized studies of radiologists and medical physicists.

Industrial radiography devices can only be used by the radiation safety officer or by a trained person with the safety officer’s written permission. If, owing to the nature of the work, the radiation safety officer cannot actively supervise compliance with the safety regulations, he or she shall appoint an operator in charge for each place where radiation is used, the operator in charge being responsible for radiation safety in the place of use and its surroundings. STUK has accepted some training organisations to give certificates of the operators in charge.

STUK oversees the training organizations accepted to give certificates.

Developing company training by making use of new pedagogical ideas and technology

Tea-Mari Latva, Antti Piirto
Teollisuuden Voima Oy

Introduction

In today's society success is based on applying new technological inventions in both training and work. From the point of view of learning and training this means that more attention needs to be paid on the ways knowledge and information technology can be used in teaching. The primary goal is to create a continuous and gradually deepening process of learning that motivates and is meaningful for the learner. The teacher should support the student in his independent knowledge structuring.

What would be the right way to approach the employees so that they would gain confidence on themselves as the guides of their own learning? What are the pedagogical ways that help the teacher in creating an efficient and a meaningful learning environment? The trainers of the technical field usually have a strong knowledge on their own area of know-how, but the pedagogical points of view have been left aside during their time of education. The main challenge is, how to integrate the theoretical know-how and the didactic skills of the trainer into a working whole?

The central terms and theories

Pedagogics cf. andragogics **=** according to the traditional concept, pedagogism means the art and education of bringing up children. Netpedagogics is a new way of learning that is based on new information and communication technology. The basis of it is a database system, which can file information in the form of text, picture or video. Andragogics is defined as a science studying adult education, which is built around the specialities of the learning situations of adults. A central goal in adult learning is continuous learning that goes on through the lifetime.

investigative learning **=** a process where the advancement of learning is guided by the problems set out by the members of the learning community, conceptions and theories formed by them and the critical evaluation of the knowledge they have searched for.

didactics **=** the central concept of didactics is education, which includes both teaching and learning. The teaching process can be divided into three phases: *1. preinteractive phase*, fore-planning the teaching, mainly realised by the teacher, but the learners can also familiarise themselves with the upcoming learning material, collect material etc. *2. interactive phase*, planning the beginning of the teaching period in co-operation with the students. The teacher and the students decide the goals, the schedule, the materials and the ways to work, after which the plan is realised and evaluated at the end. *3. post interactive phase* is for collective consideration on how to improve the teaching based on the experiences from the previous period.

constructivism **=** constructivism is the currently dominating learning concept. Previously learning was seen through behaviourism, according to the teacher–learning stimulus–student -concept. Essential aspects of constructivism are: learning by an active design of knowledge, learning is involved with action, learning is a process of interaction, a motivation to learn, to try, to solve problems and understand, the development of self-direction and the growth of ones self is possible through learning.

Education in the environment of investigative learning

Investigative learning is a new view of a learning process born from the development of intellect and thinking. In the model of investigative learning the learning and the construction of knowledge work interactively in the same way as an individual and a community.

A learning environment can be defined as pedagogically meaningful if it has *cognitive tools* to support, guide and expand the learning process. In addition, you need *tools of communication* for the co-operation between the teacher and the learner and between the learners to be possible. A net-based learning environment gives the student *tools for communal construction of information*. Digital learning materials and equipment make most of the phases of teaching work easier assuming that they are easily available and the teachers have relatively good computing skills. An essential feature of good learning material is *adaptivity*, the possibility of using it in several ways. The Internet, for example, provides you with learning material quickly, easily and extensively, the only problem is the need to select the essential and the reliable knowledge. The model of investigative learning can be divided into six components.

1) *creating a context & anchoring the teaching:* In the beginning of the learning process you need to anchor the problems you are going to deal with into the learner's earlier constructions of knowledge, *schemes*. The problem of the process of investigative learning should be complex enough and motivating for the student. In workplace training it would be important that the trainees see the solving of the problem as meaningful for their own work.

2) ***learning through problem solving:*** New knowledge is not absorbed right away, it is built by solving knowledge-problems and by creating and evaluating ones own theories and explanations. The problems that the students set for themselves have an important meaning.

3) ***explanatory learning:*** Creating a students' own working theories is another central component of investigative learning. This helps for being aware of the difference between ones own and the new conception before searching for the new information.

4) ***critical evaluation:*** The students evaluate the advancing of their study process critically by setting new goals. Critical evaluation calls for constructive interaction within a learning community so that new ideas and point of views can be found and developed. Self-evaluation is also important for the students' understanding of their own learning and themselves in control of it (*metacognition*) to develop.

5) ***acquiring new knowledge:*** The goal of an investigative learning project is to generate new understanding and knowledge. The students' working theories are tested by searching information from scientific literature, electrical sources, expert interviews and practical experiments.

6) ***sharing expertise:*** It is characteristic for investigative learning that the whole learning community is responsible for the development of the knowledge, so by mutual interaction the collective intellectual resources can be used to advance the study process, without forgetting the tutors input.

A multidimensional learning environment takes different types of students into consideration

One of the basic skills of a teacher is the skilful and meaningful use of teaching methods. A ***multidimensional*** learning environment favours diverse actions. For example in a net-based project this means differentiation of assignments so that every individual can begin their study with an assignment from their own level. This encourages and motivates the learner to develop himself level by level and creates positive learning experiences as he can see the progress of his own learning.

The autonomy of the students and the teaching groupings in a multidimensional learning context are closely connected to the idea of the *co-operation of learning*. It is essential for this concept of learning to divide the learners into heterogenical groups for the people to learn to work with different types of people. The differences between the learners can be used, among other things, to divide the people into groups where there is one expert from one area of expertise (*groups of two-way teaching*) or so that in a group the individuals are homogeneous (*expert groups*). The advantage on the first one is that different groups are equal for example in a project that requires multiple types of know-how, in the latter one the different groups could teach others their own area of expertise.

The teaching should take the *auditory*, *visual* and *audiovisual* sense channel into consideration, because different individuals use different sense channels to learn. This is especially easy to realise in a computer assisted learning environment.

Good teacher, competent instructor

Behaviouristic teaching (direct teaching) that highlights directed teaching is still a part of the whole, for example as the motivational part of the beginning. Essential for the teachers' performance is clarity and logically proceeding from the different parts to the whole.

Cognitive teaching (the processing of information) means the processing of the study material into meaningful portions where the teachers role is to be the help of the learners. Cognitive teaching is also the construction of information into flexible information structures and applying them to realistic situations.

When the previous teaching methods are connected to the technical side of information, the *humanistic* and the *social* side mainly highlight the atmosphere of the teaching. The growth into humanity happens through teaching & learning. A teachers/instructors goal in modern projects of information and communication technology is to build a "*cognitive apprenticeship*" together with the learner. Behind this concept is the idea of getting to know the culture of expertise by the more experienced expert giving the trainees a model, after which the learners take part in the processing of the more demanding assignments. The instructor guides the learners indirectly and sparingly (*scaffold*) and little by little gives them more responsibility which makes it possible for them to rise to the teachers level, possibly even above it. When you develop teaching, you should also think about the development of evaluation. A typical way of evaluation is a learning log instead of tests/examinations. The goal of the evaluation is, in addition to controlling the progress, to get feedback on your own weaknesses and strong features. This is important when developing the skills of self-evaluation. It is generally thought that a poor answer shows the stupidity of the student, when actually the answers give the teacher valuable feedback about the success of the teaching, which is the third goal of evaluation.

Research report

The purpose of this study is to find out how to develop the training of an industrial company by taking advantage of the pedagogical theories that are currently popular and the ideas about effective learning. This study will map out the opinions and the expectations connected to the training of the employees of an industrial company, when the learning environment would be based on the *model of investigative learning*. The **research problems** are:

Main problem: *How to develop the training of an industrial company in pedagogical ways?*

Sub problem 1. *What is an effective learning environment in company training in the opinion of the learners?*

Sub problem 2. *What qualities do the trainees expect and hope for from their instructor?*

Main problem: *Which and how should the principles of investigative learning be especially emphasized in the training of the different departments of different organisations?*

Research group and meters

The testees on this research all worked on the **production** and **technology** departments of the industrial company that was the research field. The enquiry was delivered to altogether 62 people, who were randomly picked from the departments mentioned before. Group managers, work leaders and other employees were chosen for the tests. From the 30 enquiries that were returned, 22 were chosen to describe the results and make the conclusions. The majority (53/62) of the people who received the enquiry were men

This research was a quantitative *survey*- research. The method used in this research was an enquiry. The enquiry attempted to outline the employees' opinions on "good training" so that effective ways could be found for the realisation of training. The respondent had to choose the best answer from prepared options, put statements into the right order or evaluate statements on the scale of 1-5 and 1-3. In addition the enquiry involved a few open questions.

Results

The results are considered from the point of view of how the employees of an industrial company have experienced training and how it could be developed according to the model of investigative learning, within a net-based learning environment.

The attached bar diagram illustrates what the research group thought about the goals of training and what was expected of and hoped for from the training. One can see from the diagram that statement 9 has received most points (40p.), when statement 5 has the least (1p.).

The goals of training

1= The training is involved with my own goals of progression in my work
2= The training is realised in a group as project work
3= Training increases my own expertise
4= The goal of the training is to guide us to effectively solve the problems we face at work
5= <u>the training helps to make you aware of the difference between your own beliefs and facts</u>
6= I can develop myself as a whole with the help of the training
7= The training makes use of all the latest technology
8= The instructor is professional and knows how to guide the learning of his students
9= <u>the training gives information of the knowledge and techniques that are needed in the work</u>

A clear majority felt that ***effective training*** is based evenly on both lectures and the exercises of the students. The people who responded thought that lecturing does not teach, active making of exercises and conversation is what teaches. Though it is good for the instructor to first present the new ideas, basic ideas, by lecturing so that the individual can begin to apply the information orally/in writing or preferably, if possible, in practise. Also the repetition of issues was considered as an essential part of the learning process.

To the question ***"where you have learned working skills"***, clearly the most common answer was in work life, in practise. It appears that basic training is only a compulsory intermediate phase for many to get into the work they want.

Effective learning environment

Multiple-choice questions were answered according to the table below. The table has been put together with the most essential results on the questions connected to learning environments.

Table 1. Considering learning environment from the point of view of effective learning.

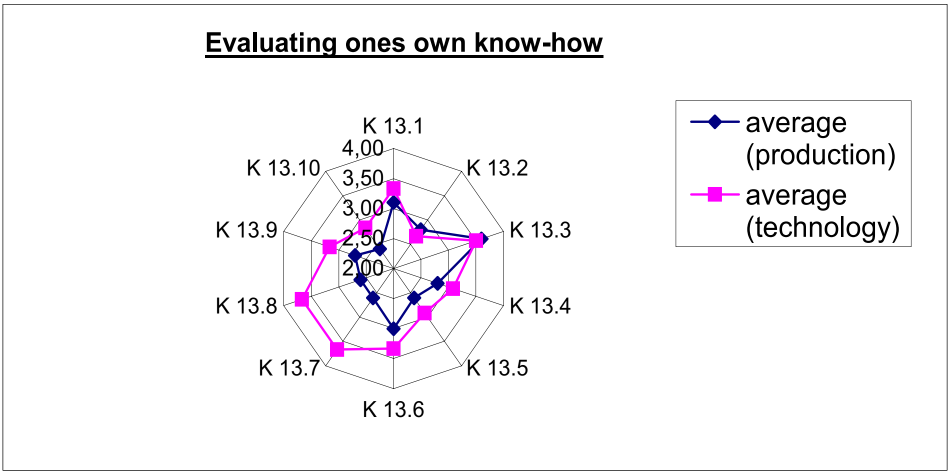
EVALUATING LEARNING ENVIRONMENT		%
1. the most suitable study situation (time&place)	does not matter not depending on time/place tied to time&place	36 27 32
2. receiving information	visually orally	86 5
3. best way of illustration	multimedia presentation video	50 5
4. best study technique	underlining writing abstracts reading text	41 32 27
5. what you learn best	practical things wholes details	50 41 5

1) It does not really seem to matter how the study situation has been organised.
2) Receiving information **visually** was clearly seen as the most effective way. Only one person saw receiving oral information as the most pleasant.
3) A **multimedia presentation** was voted as the best way of illustration as video programs were generally seen as ineffective.
4) Favourite study techniques were dependent on the person: learning is a personal thing.
5) From the issues that need to be learned the easiest to adapt were the practical things and the wholes, details were difficult to memorise.

Applying the concept of investigative learning in the training of the different departments of an organisation

The respondents had to evaluate their own know-how by giving a numerical value on the scale of 1-5 to different areas of know-how. The next picture shows a comparison between the production and technical departments' averages concerning all sectors apart.

The greatest difference between the departments is in **motivation** and **perseverance**. In the production department these qualities are evaluated as much weaker in ones on know-how than in the technical department. This result can perhaps be explained by the natural factor that the work assignments in the production department are often mechanical and short-term, when in the technical department you sometimes have to plan for long before the solution for the problem at hand becomes clear. As you can see in the picture, **working independently** and **the perceiving of the essential issues** are considered equally strong in both departments.



13.1 co-operation skills
13.2 perceiving the essential issues
13.3 working independently
13.4 ability to solve problems
13.5 ability to associate information
13.6 evaluating ones own learning
13.7 motivation
13.8 perseverance
13.9 tolerating insecurity & taking risks
13.10 applying information to practise

The respondents evaluated their computer skills on the scale of 1-5. According to the results the technical department sees their computer skills as stronger than the production department. The difference between the two departments was very even in all the sectors of computer skills, but the greatest difference was on the **control of the operating system on ones workstation**.

Only a few of the respondents had taken part in a computer assisted learning project. The experiences on them were very faint, which in itself is not a surprise because projects like that are still mostly on the development phase. Most people imagined for a computer-assisted project to demand plenty of self-discipline and own interest. Generally this type of learning was seen as **self-instruction with the help of a program**. If you were not studying manual skills, a learning method based on computers was thought to be effective, but the control of the teaching technique was considered problematic. In addition, the control of the learning results would be an absolute necessity.

Conclusions

When you consider how an industrial company could develop its training in pedagogical ways, based on this study you can say that the training should be involved with the problems and the difficult situations that have presented themselves in the work. The training motivates working if the individual feels that he can make use of what he has learned in practise. A clear majority says that they have learned the most important knowledge and skills they need in their work by "*working*". The teacher should highlight practical examples and gather wide organised wholes of the issues. These back up the deep understanding of things. Training that is only based on lectures "makes you passive and sleepy".

Only few of the respondents had been involved in a project that had applied the principles of investigative learning, but some answers showed that the thoughts involved in this model were appreciated. For example on the *good teacher* –statements the majority saw the teacher as the instructor of the construction of information and the learning as a gradually deepening research process. According to this study it would seem that the teacher is no longer considered as the informant and instead the interaction between him and the students is essential. It is important for the learner to have influence on for example the contents of the teaching and the planning of the learning environment. The main responsibility of the absorbing of the information is with the student, but the teacher is expected to be ready to guide if necessary. The goal of work place training should, in addition to the content, be the development of the individual's personality. It is clear that a motivated employee is usually more productive than a person who is bored with his work.

SESSION 10. RADIATION BIOLOGY AND EPIDEMIOLOGY

Hereditary minisatellite mutations among the offspring of Estonian Chernobyl cleanup workers

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Possible transgenerational effects of exposure to ionising radiation have been a concern since early days of radiobiological research. Despite numerous animal studies, evidence concerning the genetic effects of low radiation doses in humans remains poorly known. Previous studies have generally not revealed increased frequency of mutation rates or genetic diseases in the offspring of persons exposed to radiation. A serious limitation of earlier studies has been the low sensitivity of the monitoring systems for hereditary effects. This limitation has recently been overcome by methods based on genetic minisatellite mutations. Minisatellites are DNA regions characterised by a variable number of tandem repeats of identical 6-100 bp units. They are found with a relatively high frequency throughout the genome and exhibit a high spontaneous mutation rate. Minisatellite mutations consist primarily of gains or losses of one or more repeat units. The phenotypic effect of these mutations is unclear at present.

A single accidental event, such as the fallout released from the Chernobyl reactor 16 years ago, may expose millions of people to non-natural environmental radiation. Following the Chernobyl reactor accident, 4833 Estonian men were sent to Chernobyl to decontaminate the environment. They worked in Chernobyl between 1986-1991 and got a mean radiation dose of 110 mSv. In order to evaluate the hereditary effects of low radiation doses, we compared the minisatellite mutation rate among children born to Chernobyl clean-up workers. Minisatellite mutation rates were compared within a family, i.e. between children born before and after their father was exposed to radiation. The comparability of genetic and other factors between exposed and unexposed groups was thus maximised.

Blood samples were obtained from 147 Estonian families. The post-Chernobyl children (n = 155) were conceived within 33 months of the father's return from Chernobyl. Siblings (n = 148) born prior to the accident formed the reference group. DNA was extracted from isolated blood lymphocytes and paternity was confirmed using five minisatellite probes. Eight additional minisatellite probes were used to screen for minisatellite mutations. Mutants were identified as novel DNA fragments present in the children that could not be seen in either parent. Only paternal, independently repeatable mutations were studied, assuming that they were derived from the paternal allele closer in size.

The parental origin and germline length change were determined for 94 de novo mutations found at the eight tested loci (52 mutants among post-Chernobyl children and 42 among pre-Chernobyl children). The minisatellite mutation rate was slightly increased among post-Chernobyl children (0.042 versus 0.036 per offspring band, OR 1.33, 95% CI 0.80-2.20). Furthermore, workers with recorded doses ≥200 mSv had a three-fold mutation rate compared to those with lower doses (95% CI 0.97-9.30). The mutation rate was not associated with the father's age (OR 1.04, 95% CI 0.94-1.15), the sex of the child (OR 0.95, 95% CI 0.50-1.79) or the time interval between the birth of the child and the father's return from Chernobyl (OR for children born ≥49 weeks after the return date was 1.45, 95% CI 0.84-2.52).

Conflicting results have been obtained on radiation-induced minisatellite mutations in humans. No increase in minisatellite mutation rate has been detected in the offspring of atomic bomb survivors in Hiroshima and Nagasaki[1-3], in human sperm following radiotherapy[4], or in children of Chernobyl clean-up workers in Ukraine[5-6]. Yet, roughly doubling of the minisatellite mutation frequency has been reported among children in Belarus[7-8] and around the Semipalatinsk nuclear test site in Kazakhstan[9].

In this study, maximised comparability was achieved by comparing the children born before and after paternal radiation exposure within the same family. We found a slight, non-significant increase in the minisatellite mutation frequency in post-Chernobyl children when compared to their pre-Chernobyl siblings. However, workers with doses ≥200 mSv had a three-fold increased mutation rate. Our findings suggest that genetic effects may occur at lower doses than has previously been assumed.

The full study will be published in Radiation Research in 2003[10].

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Minisatellite mutations and retrospective biodosimetry of population living close to the Semipalatinsk nuclear test site

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During the period between 1949 and 1989 nuclear weapon testing carried out at the Semipalatinsk Nuclear Test Site (STS) resulted in local fallout affecting the residents of Semipalatinsk, East Kazakhstan and Pavlodar districts of Kazakhstan and Altai region of Russia. The Semipalatinsk nuclear polygon in Kazakhstan has been the site for 470 nuclear tests, including 26 tests performed on the ground and 87 in the atmosphere. More than 1.5 million people living in the vicinity of the test site were repeatedly exposed to ionizing radiation.

The objectives of the study were: 1) to establish a biosample database of blood samples of families in three generations living close to the STS and control families in three generations from clean areas, 2) to determine the minisatellite mutation rates in the three generations of exposed people and the control families of the same ethnic origin living in non-contaminated areas, and 3) to determine the chromosomal translocation frequencies by FISH chromosome painting in the lymphocytes of the exposed and the control people in order to determine the radiation exposure. The aim of the study was to select the population living near to the STS and subjected to the greatest radiation exposure. Of particular interest was the first test of 29th of August 1949, as this was reported to have caused heavy fallout along a narrow trajectory extending north-east from the Polygon, also covering parts of the Altai region of Russia and parts of Pavlodar and Karaganda regions in Kazakhstan. This first explosion occurred at an altitude of 30 m above ground with an energy release of 22kT. The wind velocity at the time of the test was 45-50 km/h, and within 2 hours a radioactive cloud reached densely populated areas inside a 100 km radius from the hypocentre. Population living in the villages of Dolon, Bodene, Cheremushki, Mostik, Kanonerka,Chagan and Karamyrza settlement (close to Kanonerka) of the Beskaragai District of Semipalatinsk region was selected for the study.

The subjects available for the study (three-generation families) were defined by a feasibility study identifying altogether 1029 persons in 83 families living in 7 villages in the Beskaragai district of the Semipalatinsk region. These villages were affected by the fallout from the first nuclear test in August 1949. Families for the final study were selected according to preset criteria ensuring that the grandparent generation was exposed at the time of the nuclear test, that their children were conceived after the main exposure (born after September 1950), and that there was an adequate number of family members available for genetic analyses. Finally, members of 40 families (361 individuals) were interviewed and sampled.

The inhabitants of Dzerzhinsk, Zhanatalap and Ushtobe villages of former Taldy-Kurgan District were included in the study as a control group. After careful interviews and matching (age, ethnic origin, socio-economic factors) of the controls, a group of 28 control families (total 250 people) were chosen. Background data on family, residential history, radiation exposure, ethnicity, age, gender, smoking habit and lifestyle of all studied families were recorded using a questionnaire to all members of both cohorts, and controls were matched by these characteristics.

Minisatellite mutation analysis

The analysis of germline mutations at tandem repeats minisatellite loci has previously been used to evaluate genetic consequences of exposure to ionizing radiation in humans and mice. The results of these studies have shown that, in sharp contrast to traditional genetic systems, radiation-induced changes in mutation rate can be detected in very small population samples and following a relatively low-dose exposure to ionizing radiation. Using this approach, germline mutation rates were estimated in the exposed cohort of families from the rural area near the Semipalatinsk nuclear test site and in a control group of families of the same ethnic origin living in non-contaminated areas. The study has been published in 2002 (Dubrova et al. 2002).

Briefly, blood samples were collected from 40 three-generation families (parental year of birth 1926-1948) inhabiting the rural areas of the Semipalatinsk District and from 28 three-generation control families (parental year of birth 1920-1951) inhabiting the rural area of the Taldy-Kurgan District. All families were profiled using eight hypervariable single-locus minisatellite probes B6.7, CEB1, CEB15, CEB25, CEB36, MS1, MS31 and MS32, chosen for their relatively high spontaneous mutation rate. Minisatellite mutation analysis was performed on samples from 516 persons. Parental allele sizes and allele-length frequency distributions were indistinguishable between the control and irradiated families. However, minisatellite mutation rate in the cohort of P_0 parents directly exposed to radioactive fallout from the surface and atmospheric nuclear tests was 1.8-fold higher than in the control non-exposed population from the Taldy-Kurgan District. Less marked 1.5-fold increase was also found in the F_1 parents from the affected area. Most importantly, minisatellite mutation rate in the cohort of F_1 parents from the affected area showed a significant negative correlation with the year of birth, consistent with the decay of radioisotopes after the cessation of surface and atmospheric nuclear tests. Despite an elevated mutation rate in the exposed group, the size spectrum of mutants in control and exposed group were similar. The results of our study provide support for our previous findings showing an elevated mutation rate in the families exposed to the post-Chernobyl radioactive fallout (Dubrova et al. 1996, Dubrova et al. 1997) and are important for the further studies of genetic risks of ionizing radiation for humans.

Retrospective dosimetry using FISH translocations

Translocation analysis using FISH (fluorescence *in situ* hybridization) chromosome painting was performed to evaluate the magnitude of cumulative exposure to ionizing radiation among the population living close to the Semipalatinsk nuclear test site in the Beskaragai district in Kazakhstan. The study has been published in 2002 (Salomaa et al. 2002). Altogether 60 persons, all males, were selected from the cohort living in seven villages in the area affected by the fallout and 40 persons, all males, from the control cohort living in non-contaminated areas. In both groups, approximately half of the subjects represented the P_0 generation. Individuals for the control cohort were chosen to match the exposed cohort with respect to, among others, age, smoking and ethnic background.

FISH analysis of almost 2000 metaphases per subject was performed. Similar translocation frequencies were observed in the Semipalatinsk cohort (7.0 per 1000 cell equivalents C.E.) and in the matched control group (7.9 per 1000 C.E.). In the P_0 generation, i.e. individuals who lived in the area of radioactive fallout from the first nuclear test in August 1949, translocation yields were almost equal to the corresponding controls (10.0 and 10.2 per 1000 C.E., respectively). Neither was there a difference in the mean translocation frequencies between the various test villages. Assuming translocation stability in peripheral blood lymphocytes over several decades, these findings suggest that on average, the magnitude of exposure to the studied cohort in the

Semipalatinsk area has been considerably smaller than that reported in the literature (Gusev et al. 1997). Previously reported doses in the order of 1-4.5 Gy cannot be confirmed by the present data. In the multiple regression analysis performed to evaluate the effect of various confounders, only age had statistically significant relationship to the translocation frequency. Due to the uncertainties involved, a positive dose effect can be considered to be present if doubling of the age-dependent control level is observed. In the P_0 group studied here, using the linear coefficient of the dose effect curve, which is valid in chronic exposures, a minimum average dose of 0.5 Gy, mainly received through external exposure, would have been reliably detected by the technique. Thus, we conclude that the studied subjects, who lived in the area of radioactive fallout in 1949, have received cumulative doses of less than 0.5 Gy.

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Posters

10a Delayed changes in gene expression in human fibroblasts after alpha irradiation

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Introduction

It has been commonly accepted that the biological consequences following radiation exposure are attributable to DNA damage and expressed within one or two cell generations. Recent evidence, however, has now been emerged to challenge this classical paradigm. Changes in non-irradiated bystander cells may lead to transmissible genomic instability. This phenomenon has been termed "non-targeted" and in addition to genomic instability, includes also radiation-induced bystander effects. Various types of genomic damage can be observed in affected cells for many generations after irradiation. After alpha-particle irradiation, delayed non-clonal chromosomal aberrations were seen in surviving cells of cultured haematopoietic stem cells from CBA/H mice. These aberrations were mostly of non-identical chromatid type, showing that they had arisen for many generations after the irradiation (Kadhim et al. 1992).

Although radiation-induced genomic instability has been observed in several in vitro and in vivo experiments, the mechanisms involved in the induction and transmission of genomic instability remain unknown. The purpose of this work was to provide new information about the delayed or persistent effects of radiation on expression of genes associated with chromosomal instability phenotype. It has been assumed that this phenotype is linked to sustained alterations in gene expression rather than to specific gene mutations. The delayed gene expression changes in cells after irradiation have not been extensively studied. Human syndromes expressing chromosomal instability have been demonstrated to have a role in the evolution of malignancy. Thus, the role of radiation-induced genomic instability in radiation oncogenesis is of importance. The work is part of the joint EU-funded project called "Genomic instability and radiation-induced cancer" (RADINSTAB). The aim of the RADINSTAB project was to investigate health effects of genomic damage, predisposition to cancer and correlation of genomic instability endpoints with radiation-induced cancer.

Gene expression changes in human fibroblast cells at delayed time points after alpha particle irradiation were studied. The aim was to identify genes playing pivotal role in inducing genomic instability.

Materials and methods

A normal human fetal fibroblast cell line, designated HF19, was used. These cells have been previously shown to express radiation-induced chromosomal instability (Kadhim et al. 1998). Before irradiation, the height of the cell monolayer was determined with confocal microscopy in order to ensure the ability of alpha particles to traverse the monolayer. HF19 cells were irradiated with a dose of 0.375 Gy and 0.75 Gy (figure 1).

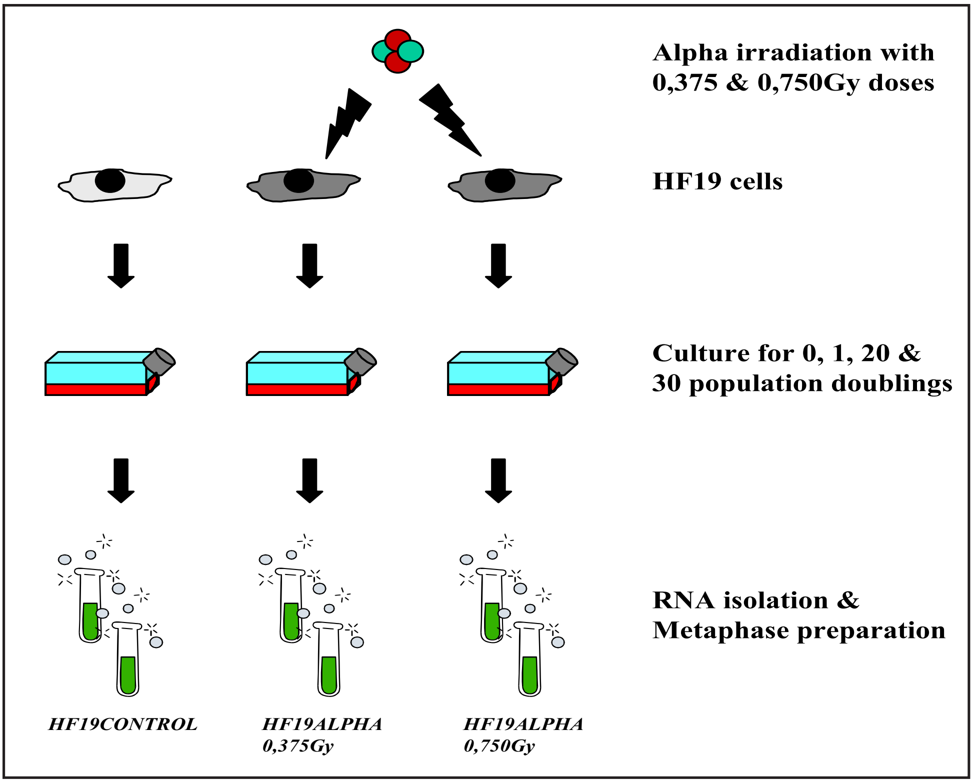

Figure 1. The experimental setup.

Cells were allowed to go through 0, 1, 20 and 30 population doublings after irradiation. The first sample was taken 2 hours after irradiation, the second 3 days, the third 38 days and the fourth 58 days after irradiation. Cells irradiated with 0.75 Gy were used only at the first two time points. At each time point part of the cells were harvested and RNA was isolated. Metaphase preparations were made at all time points except the first. The gene expression changes were studied using cDNA array hybridisation method (Clontech Atlas™) and confirmations were made with semi-quantitative RT-PCR method. The expression of cancer-related genes was investigated with cDNA array hybridisation method. Genes of approximately twofold difference in mRNA expression levels (measured as normalised intensities of dots) were accepted for RT-PCR confirmation. Semi-quantitative RT-PCR was performed using gene specific primers. Two housekeeping genes were amplified in parallel to the genes of interest. Samples were taken between 18-38 cycles of PCR and run on agarose gel. The normalised intensities of the bands were compared and genes of approximately twofold difference in intensities were considered as confirmed gene expression changes. The metaphase preparations were analysed by solid staining and mFISH for chromosomal instability.

Results

The results of chromosome analysis by solid staining method showed transmissible chromosomal instability in cells irradiated with a dose of 0.375 Gy. The mFISH analysis is in progress. The data shows that 0.375 Gy α -particles induced transmissible chromosomal instability as defined by non-clonal chromosomal aberrations with a high frequency of chromatid type aberration. The substantial number of non-clonal chromosomal aberrations in irradiated cells studied at delayed times after irradiation confirmed the induction of chromosomal instablility also in this experiment.

Analysis of cDNA array showed upregulation in 22 genes and downregulation in 9 genes, most of them at the early time points. With cDNA array, upregulation of CDKN1A (p21/Waf) gene was detected at early time points after irradiation with both doses. These findings were re-examined with semi-quantitative RT-PCR method. RT-PCR confirmed upregulation of five genes and downregulation of two genes. Six of the confirmed changes were found in samples irradiated with 0.75 Gy, taken at the first two early time points after irradiation. Only one of the changes was in a sample irradiated with 0.375 Gy, taken at the first time point. Two hours after irradiation, the CDKN1A (p21/Waf) was upregulated in cells that had received 0.75 or 0.375 Gy.

This gene was expected to be upregulated after irradiation, since it is normally induced after many kinds of cellular damage. p21/Waf protein can induce G1 arrest and block entry into S phase. Overexpression of CDKN1A results in G1 arrest and has been shown to suppress effectively tumor growth in vitro and in vivo. Increased expression of Hexabrachion (HXB) in cells irradiated with 0.75 Gy was detected at two hours after irradiation. This protein is an important extracellular matrix hexameric glycoprotein, which is selectively expressed at sites of tissue remodeling in developing and pathological connective tissues. A high Hexabrachion expression has been found during carcinogenesis in almost all organs. Hexabrachion increases rapidly after inflammation or injury, so it was no surprise to observe upregulation of this damage responsive gene also after irradiation. Matrix metalloproteinase 11 gene (MMP-11) was upregulated in cells irradiated with 0.75 Gy three days after irradiation. This protein is a member of the family of metalloproteinase enzymes, which degrade the extracellular matrix. MMP-11 production has been associated with malignant phenotype, tumor invasion and metastasis. Increased expression of MMP-11 in HF19 cells may participate in an early step towards malignancy. Another gene expression change observed three days after irradiation was downregulation of a DNA excision repair protein, ERCC1, in cells that had received 0.75 Gy. This protein is involved in the early steps of the excision of DNA damage; it serves as an excision nuclease. Decreased expression of a repair gene in HF19 cells may be one of the initiating steps in destabilisation of the genome.

Conclusions

Our results show that radiation-induced changes in gene expression were detected in cells irradiated with 0.75 Gy of alpha-particles. Upregulation of cell damage and stress induced genes were seen two hours after irradiation. The changes in expression of carcinogenesis related genes and in a repair gene already three days after irradiation may play a role in events leading to destabilisation of the genome. Gene expression changes were observed only relatively soon after irradiation and mainly in cells irradiated with the higher dose. Changes in mRNA levels could not be detected at delayed time points. These data indicate that changes in gene expression, in general, may be most reliably detected in samples irradiated with high doses. A notable amount of cells expressing chromosomal instability was observed at delayed time points among the cells irradiated with the lower dose; thus detection of genes responsible for maintaining this instability would have been expected. According to this study there seems not to be any specific changes in expression of any gene that could be shown to be responsible for maintaining the chromosomal instability at delayed times after alpha-particle irradiation. Investigation of chromosomally unstable cell clones could perhaps provide more data on gene expression changes related to genomic instability. Unfortunately, production of cell clones was not possible in the scope of this study.

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10b Radiation exposure and cancer incidence among a Finnish group residing in the city of Kiev during the Chernobyl accident

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Introduction

The worst nuclear accident in the world so far took place April 26, 1986, in Chernobyl, Ukraine. At the time of the accident there was a group of Finnish construction workers in the city of Kiev, situated about 100 km south from the nuclear power plant. Since there was a concern about the possible health effects of the accident, the radiation exposure of this group was measured and a cancer analysis was conducted.

Materials and mehtods

The total number of subjects in the exposure assessment and cancer analyses was 139 (123 men and 16 women), mostly middle-aged male construction workers. The radiation exposure assessment of this cohort was retrospectively based on three different methods: film dosimetry, whole-body counting and chromosome analysis. They were performed during 1986 and 1987 both in Kiev, Ukraine and in Helsinki, Finland. The background information of the study population was collected from the Population Register Centre, and all cancer cases in the cohort from April 26, 1986 to December 31, 1999 were collected from the Finnish Cancer Registry. The actions for radiation exposure assessment were conducted by STUK and the cancer analysis by the Finnish Cancer Registry.

Results

The radiation doses among the study population were minor. According to the film dosimetry the mean external dose was 3.3 mSv and according to the whole body counting measurements the mean internal dose and the mean thyroid dose were 0.5 mSv and 11 mSv, respectively. The chromosome analyses showed no differences from normal background values. The mean frequency of dicentric chromosomes was 0.0008, and that of total chromosome aberrations was 0.005. The total observed cancer incidence in the study population was, however, higher than expected (SIR 2.69, 95% CI 1.16-5.30).

Conclusions

The radiation exposure of the Finnish group was quite insignificant but the cancer incidence in the study population was nevertheless higher than expected. Since the study population was so small and the observed cancer types not usually associated with radiation exposure, the role of chance and bias needs to be carefully considered.

10c Occupational Radiation Exposure to Low Doses of Ionizing Radiation and Female Breast Cancer

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Abstract

The aim of this study is to examine the relationship between past occupational radiation exposure to low doses of ionizing radiation and cases of diagnosed and registered breast cancer [probability of causation - PC] among Bulgarian women who have used different ionizing radiation sources during their working experience.

The National Institute of Health (NIH) in US has developed a method for estimating the probability of causation (PC) between past occupational radiation exposure to low doses of ionizing radiation and cases of diagnosed cancer. We have used this method.

A group of 27 women with diagnosed breast cancer has been studied. 11 of them are former workers in NPP – "Kozloduy", and 16 are from other sites using different sources of ionizing radiation. Analysis was performed for 14 women, for whom full personal data were available. The individual radiation dose for each of them is below 1/10 of the annual dose limit, and the highest cumulative dose for a period of 14 years of occupational exposure is 50,21 mSv.

The probability of causation (PC) values in all analyzed cases are below 1%, which confirms the extremely low probability of causation (PC) between past occupational radiation exposure to low doses of ionizing radiation and occurring cases of breast cancer.

Keywords: probability of causation (PC), breast cancer, occupational radiation exposure, low doses of ionizing radiation

Introduction

The statistical data show that the incidence of cancer is increasing nows days. At the head of the list in Bulgaria as well as in the whole world is breast cancer. This is demonstrated by trends of cancer incidence among white women in the US, 1947 – 1985 (figure 1) /1/ and in Bulgaria, 1981 – 1995 (figure 2) /2/.

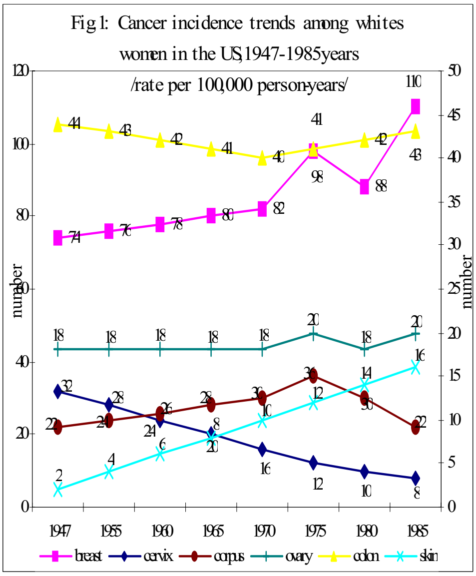


Fig 1: Cancer incidence trends among whites women in the US 1947-1985 years /rate per 100,000 person-years/

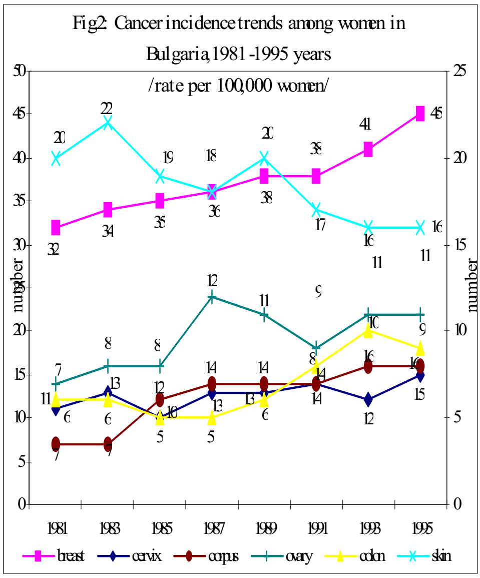


Fig 2: Cancer incidence trends among women in Bulgaria, 1981-1995 years /rate per 100,000 women/

It is well known that a great number of endogenous and exogenous factors can be connected with breast cancer. Among them the occupational factors are of great importance, and many researchers study them using different methods: administrative, cohort and case-control studies (figure 3).

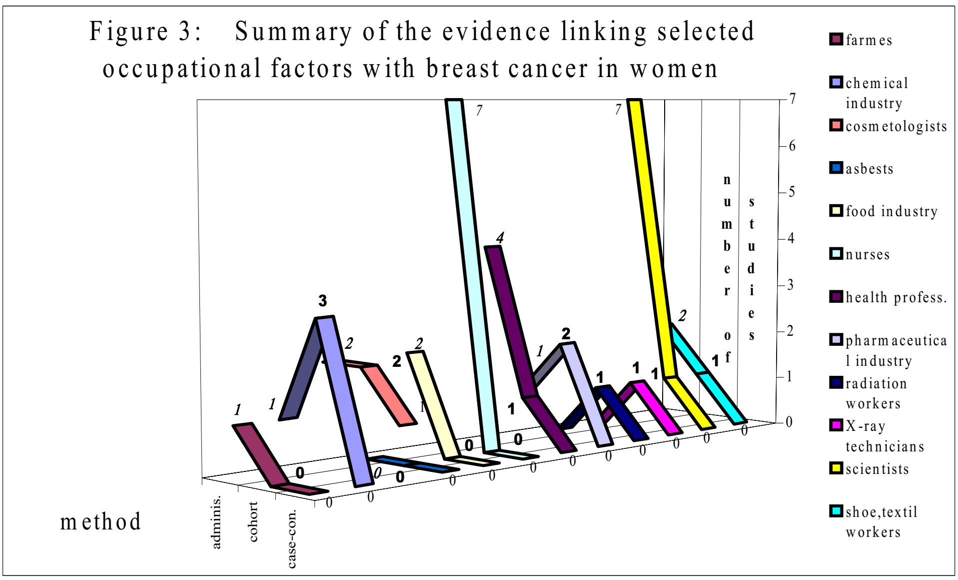


Figure 3: Summary of the evidence linking selected occupational factors with breast cancer in women

The aim of the study

The aim of this study is to examine the relationship between past occupational radiation exposure to low doses of ionizing radiation and diagnosed breast cancer [probability of causation – PC] among Bulgarian women who have used different ionizing radiation sources during their working experience.

Materials and Methods

An ad hoc Working Group in NIH has developed a method, based on radioepidemiological tables, for estimating the probability of causation (PC): the relationship between past radiation exposure and a specific cancer /4/. We applied this method in our study.

The probability of causation (PC) is defined as a part of risk for cancer disease in different age groups. The probability of causation (PC) is expressed by the relation between the increase of the probability of development of cancer in case of radiation exposure and the actual probability of this cancer to be developed at similar conditions. The probability of causation (PC) takes into account the role of different characteristics for both probabilities. The Relative Excess Risk (R) has been introduced to facilitate the calculation of probability of causation (PC). The Relative Excess Risk (R) is a product of several numbers, which we can take from tables or we can calculate them easily.

$$R = F/D/ \, . \, T/Y/ \, . \, K/A,S/ \qquad /1/$$

where: F/D/ expresses the influence of dose of ionizing radiation on the risk;

T/Y/ expresses the variation of the risk according to the latent period;

K(A,S) expresses the changes of the risk related to sex (A) and age(S) at the time of exposure

The occupational radiation exposure is usually repeated during the whole working experience. We have accepted that the Relative Excess Risk (R) for the whole period of exposure is a sum of annual risks because there is no more precise method. The confidential interval is 90% for each of the calculated probability of causation (PC) values.

$$PC = \qquad\qquad\qquad /2/$$

We need definite background personal data for the estimation of probability of causation (PC): the date of birth, the year of breast cancer diagnosis, the individual radiation dose per year for the whole period of exposure. We have taken these doses from Dozimetry Unit in NPP – "Kozloduy" and from laboratory "Dozimetry of external exposure" in the National Center of Radiobiology and Radiation Protection.

In this study have been included women who have met both conditions: diagnosed and registered breast cancer and past occupational radiation exposure. The women in the study have had different professions: workers in NPP – "Kozloduy", radiologists, technicians in radiology, chemists and a bookkeeper.

27 women meeting both conditions have been registered in the Department of Radiation Medicine and Epidemiology in the National Center of Radiobiology and Radiation Protection. We have had full background personal data only per 14 of them and only these have been included in the study.

Results

We have ascertained the fact that the personal radiation exposure for each of the cases is below 1/10 of the annual dose limit and the highest cumulative dose for a period of 14 years of occupation is 50,21 mSv. The calculated probability of causation (PC) values with 90% CI are showed in table 1 and on figure 4.

Table 1. The calculated PC with 90% CI .

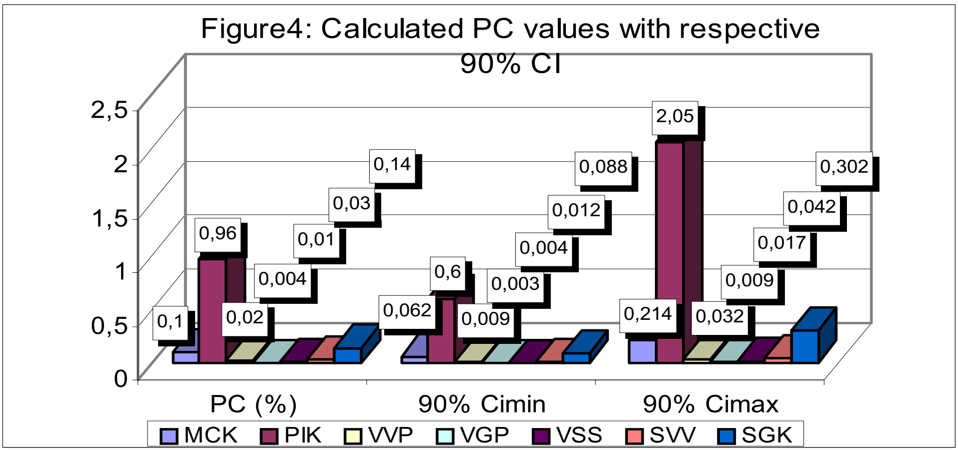
Initials	PC (%)	90% CI	Initials	PC (%)	90% CI
AIB	0,00	0,00	VGP	0,004	0,003 – 0,009
VVF	0,00	0,00	VSS	0,01	0,004 – 0,017
MCK	0,10	0,062 – 0,214	DPV	0,00	0,00
PIK	0,96	0,600 – 2,050	EGT	0,00	0,00
SSC	0,00	0,00	SVV	0,00 /right/	0,00
AGV	0,00	0,00		0,03 /left/	0,012 – 0,042
AOS	0,00	0,00	SGK	0,14	0,088 – 0,302
VVP	0,02	0,009 – 0,032			

We can not give an exact estimation of the relationship between past occupational radiation exposure to low doses of ionizing radiation and cases of diagnosed breast cancer at this stage of knowledge for risk from ionizing radiation.

The reasons are as follows:

1. There are a lot of inexactness in radioepidemiological studies;

2. Many extrapolations are made and they are not always correct: extrapolation of risk from higher to lower doses and dose rate of radiation; extrapolation from one population to an other population without taking into account their specific characteristics;



The probability of causation (PC) for each of the investigated cases is below 1%.

3.The method allows the risk to be estimated for lifetime, but exposed populations have never been followed till death;

4.This method does not take into account most of the individual characteristics although they influence radiosensitivity;

5.This method does not take into account the total effect of a combination of several cancerogenic factors.

Conclusions

1. Radiation dose for each case is below 1/10 of the annual dose limit;
2. The highest cumulative dose for a period of 14 years of occupation is 50,21 mSv.
3. The probability of causation (PC) for each of the investigated women is below 1%;
4. We do not find a relationship between past occupational exposure to low doses of ionizing radiation and cases of diagnosed and registered breast cancer [probability of causation – PC] among Bulgarian women who have used different ionizing radiation sources during their working experience.

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SESSION 11. RADIOECOLOGY AND MONITORING

Gaussian plume model analysis of atmospheric dispersion measurements of radioactive releases from the BR1 research reactor in Mol, Belgium

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Abstract
Data from an atmospheric dispersion experiment using a visible tracer along with the routine releases of ^{41}Ar from the BR1 air-cooled research reactor in Mol have been analysed in terms of a Gaussian plume model. Measurements of plume geometry and of the radiation field from the ^{41}Ar decay are compared to model calculations and good overall agreement is found between model and experiment.

Introduction
Recently, a full-scale atmospheric dispersion experiment was performed at the BR1 research reactor at the Belgium Nuclear Research Centre (SCK•CEN) [1]. The experiment was carried out as collaboration between Nordic Nuclear Safety Research (NKS) and SCK•CEN, with participants from Risø National Laboratory, SCK•CEN, Danish Emergency Management Agency (DEMA) and the Technical University of Denmark (DTU). The aim of the experiment was to obtain data characterizing radioactive releases from a nuclear installation with the purpose of testing atmospheric dispersion and dose rate models.

In the experiment a visible tracer was released from the 60-m reactor stack along with the routine emissions of argon-41 from the air-cooled reactor. Simultaneous measurements of the plume characteristics were performed including source term, meteorology, aerosol plume geometry and downwind radiation field from the decay of ^{41}Ar. The measurements covered approx. 6 hours of continuous reactor operation spanning three days during the period October 1-5, 2001, with the remaining two days used for set-up and calibration. A further 6 hours of radiation data were obtained without the aerosol tracer measurements. The main wind advection direction during the measurements was more or less constant, the wind speeds low and the estimated atmospheric turbulence predominantly stable to neutral.

The radiation measurement data have been analysed and arranged in a database, that can be used to test atmospheric dispersion and dose rate models but also in developing data assimilation models for atmospheric dispersion of radioactive releases. The present paper describes the analysis of atmospheric dispersion and ^{41}Ar radiation field data based on a Gaussian plume model.

Measurements

During the experiment, the reactor output was kept at 700 kW. At this effect atmospheric air is led through the reactor at a rate of 9.4 $m^3 s^{-1}$ giving rise to a constant ^{41}Ar emission rate of approx. 1.5 x 10^{11} Bq h^{-1}. The ^{41}Ar activity concentration inside the stack is recorded continuously using a plastic scintillator mounted with a photo multiplier.

Meteorological observations of wind speed and direction, temperature and precipitation were performed by an array of permanent instruments mounted on the weather mast of SCK•CEN. Observations were recorded every minute and subsequently used to estimate the dispersion scaling parameters for 10-minute intervals using the micro-meteorological pre-processor by Astrup et al. [2].

The geometry of the radioactive plume in the experiment was determined using a Lidar scanning technique [3]: a white aerosol tracer was injected into the stack at about 30 m height and the aerosol plume emitted from the top of the stack assumed to be well mixed with the argon plume. Crosswind plume profiles were determined by scanning the plume with a pulsed laser beam, using a high-resolution aerosol-backscatter Lidar system. Aerosol positions are determined from the time delay of the echo, the strength of which is related to the aerosol particle concentrations in the plume. From the cross-section measured, the plume dispersion coefficients (σ_y, σ_z) were derived along with the mean position and elevation of the plume and made available for this study [4].

The radiation field was monitored using two arrays of four NaI(Tl)-detectors supplemented by two high-resolution germanium detectors. One array of NaI(Tl) detectors provided by DEMA had thermally insulated 3" x 3" crystals and recorded 512-channel energy spectra every 30 seconds while four detectors provided by SCK•CEN consisted of non-insulated 2" x 2" crystals connected to a single-channel counter yielding an integrated count rate every minute. The NaI(Tl)-detectors were deployed along two lines perpendicular to the main wind advection direction at distances up to 1,500 m from the stack – in one case the detectors were grouped two by two along a single line for calibration purposes. On each line, the detectors were placed approximately 100 m apart.

Using the spectral radiation measurements the natural background for each detector position was estimated and the background-subtracted full-energy-peak count rate for ^{41}Ar decay at 1293.6 keV derived. From the net window count rate, n, the fluence rate was obtained as

$$\phi = \frac{n}{\varepsilon},$$

where ε is the mean detector efficiency (the peak response). The mean detector efficiency was determined from a Co-60 calibration experiment. For the non-insulated NaI(Tl) detectors, background count rates and detector efficiencies were determined from the inter-calibration experiment with the DEMA NaI(Tl) detectors.

The germanium detectors were mounted along the plume centreline or next to the NaI(Tl) detectors, providing further calibration measurements. From the germanium measurements it was determined that the emissions from the reactor contained no measurable traces of other radioactive isotopes apart from ^{41}Ar [5].

Gaussian plume model analysis

The measurement results have been compared to the Gaussian plume model for atmospheric dispersion. In this model, the ^{41}Ar activity concentration is given by

$$\chi(x,y,z) = \frac{Q(x)}{2\pi u \sigma_y \sigma_z} \exp\left(-\frac{y^2}{2\sigma_y^2}\right)\left\{\exp\left[-\frac{(z-h)^2}{2\sigma_z^2}\right] + \exp\left[-\frac{(z+h)^2}{2\sigma_z^2}\right]\right\}$$

where Q is the source term, σ_y and σ_z the dispersion coefficients, u the wind speed, and h the effective release height. The coordinate z is the height above the ground and y is the horizontal distance from the plume centreline. The fluence rate is obtained as

$$\varphi(\vec{r}_0) = \frac{1}{4\pi}\int d^3r \, \frac{\chi(\vec{r}) \cdot e^{-\mu|\vec{r}-\vec{r}_0|}}{(\vec{r}-\vec{r}_0)^2}$$

where μ is the linear attenuation coefficient in air.

The model plume dispersion coefficients (σ_y, σ_z) are obtained using a modified power law representation of the Pasquill-Gifford stability curves [6]. To account for an initial vertical dispersion close to the release point, a term σ_{z0} was added quadratically to yield the effective vertical dispersion,

$$\sigma_{z,eff} = \sqrt{\sigma_z^2 + \sigma_{z0}^2}$$

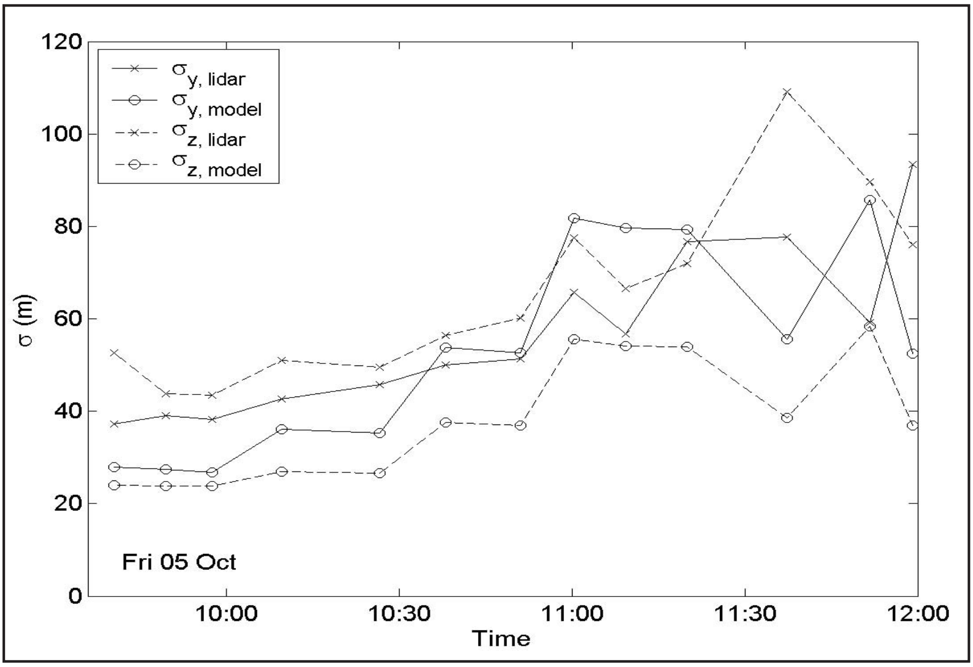

Fig. 1. Dispersion coefficients from Lidar measurements and from Gaussian plume model at a distance of approximately 400 m from the stack. The increase of the dispersion coefficients with time reflects increasing turbulence during the experiment.

In Fig. 1, the atmospheric dispersion parameters obtained by the Lidar measurements for Friday Oct. 5 are compared to the model results, using σ_{z0} = 20 m. Good agreement between the measured values and the model results is observed, both with respect to the horizontal and vertical dispersion. Similar results were obtained for the measurements Wednesday and Thursday.

The fluence rate was calculated within the Gaussian plume model using an asymptotic approximation [7]. The results obtained for Friday are shown in Fig. 2 along with the measurements. Again, the model calculations are seen to reproduce the overall behaviour of the data, the experimental results, however, exceeding the model by roughly a factor 1.5-2.

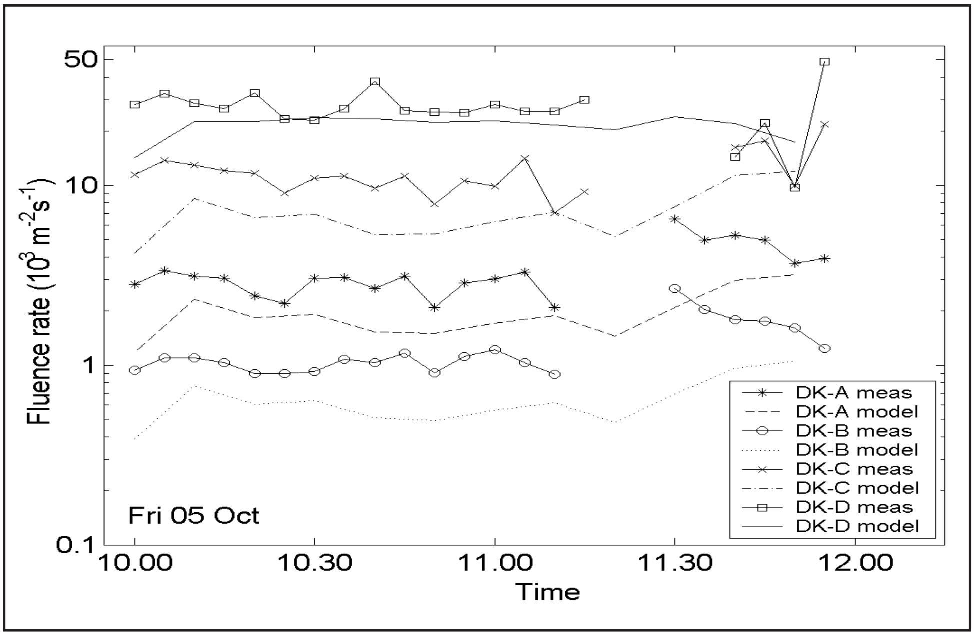

Fig. 2. Calculated fluence rates compared to measurements. The NaI detectors DK-D, DK-C, DK-A, DK-B are placed approx. 200 m from the stack and 0 m, 100 m, 200 m and 300 m from the plume centreline, respectively.

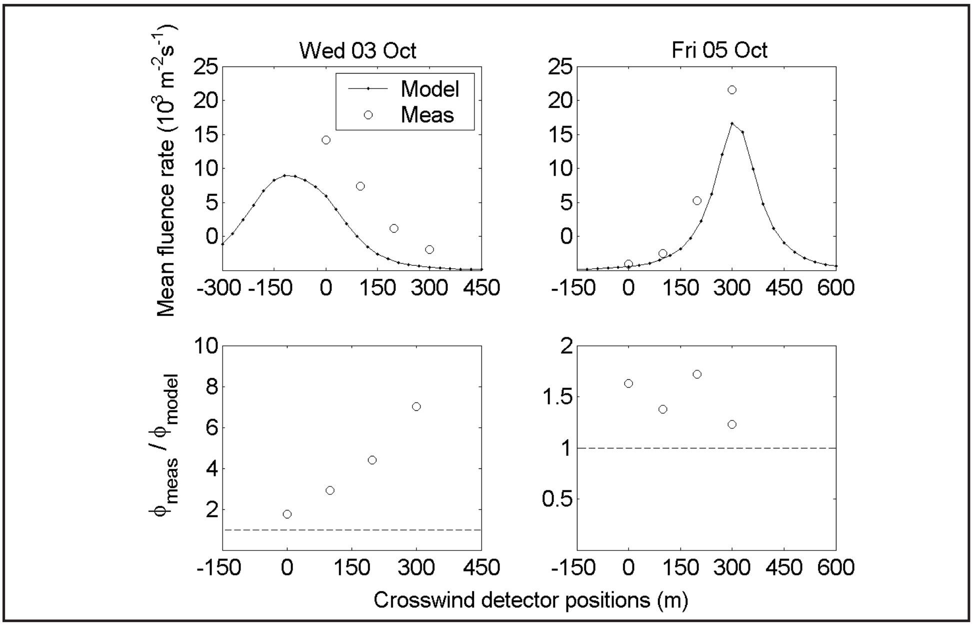

Fig. 3. Gaussian plume model calculations compared to mean crosswind fluence rates measured on Oct. 3 and Oct. 5 over a period of 40 and 90 minutes, respectively. In bottom panels the ratio $\varphi_{meas}/\varphi_{model}$ are shown.

Fig. 3 shows the mean fluence rates obtained for Wednesday and Friday. The model results shown in the top panels in both cases follow the bell-shaped crosswind form expected from the Gaussian plume model. The distribution observed for Wednesday is wider than the distribution observed Friday, because the detectors were placed approx. four times as far away from the stack. In both cases, the ratio $\varphi_{meas}/\varphi_{model}$ between measured and calculated fluence rates grows with the distance from the plume centreline, more pronounced on Wednesday (left bottom) than Friday (right bottom). This indicates that the actual crosswind distribution is wider than given by the standard Gaussian plume model.

Conclusions

Measurement data from an atmospheric dispersion experiment at the BR1 research reactor in Mol, Belgium have been analysed and arranged in a database suitable for evaluation and development of atmospheric dispersion and dose rate models for nuclear emergency preparedness.

Gaussian plume model calculations have been compared to the measurement data, and reproduce well both the atmospheric dispersion parameters measured by Lidar scanning, and the gross radiation field measurements. The measured fluence rates, though, are found on average to exceed the model predictions by a factor 1.5-2, or even more at large distances from the plume centreline.

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The radiation environment in western Sweden

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Introduction

In case of an accidental release of radionuclides it is important to be able to make a fast, but reliable, estimation of both the radiation dose to people in the affected area and the extent of the fallout. These estimations may be made with greater accuracy if background ambient dose levels and concentrations of radionuclides in the ground, within the particular area, are known prior to the fallout. Such background measurements are preferably performed at predetermined (reference) sampling sites. A frequent monitoring of background data at the reference sites will also make it possible to follow variations in the concentration of anthropogenic, as well as, naturally occurring radionuclides in the environment and hence to study the radiation dose to people and biota.

The aim of this project was therefore to survey the radiation environment in western Sweden by

soil sampling, field gamma spectrometry and intensimeter measurements at pre-chosen reference sites for collection of background data estimating the background radiation doses to the inhabitants by TLD worn by a group of people living in the area

Materials and methods

Reference sites

In this work, western Sweden is defined as the area within approximately 100 kilometres from the city of Göteborg. In this area, 25 communities were chosen as *reference communities*, due to their relatively large population. In each reference community, a *reference site* for field measurements and soil sampling was chosen in the central part of the community. All the sites were plane, grass-covered surfaces, as large as possible. In the reference community Göteborg, the largest community in the examined region, 6 reference sites were chosen which had all previously been used as reference sites for in situ measurements and soil sampling (Edbom, 1996; Ghiasi, 1997; Rask, 1999; Standar, 2002).

At all reference sites, an *in situ* measurement, using a HPGe detector (efficiency 10 % and resolution 1.9 keV @ 1332 keV) connected to a multi channel analyser and a laptop computer, was performed. The detector was placed 1 m above the ground, looking downwards and the sampling time for the acquired spectra was 1000 s each. The *in situ* measurements were performed at the same locations as the soil sampling.

The soil samples were collected with a cylindrical corer of 8 cm diameter. At each site three cores were taken, sliced in 2 or 3 cm layers, down to a depth of 15 cm. The soil from corresponding layers in each core was then combined to one sample (Isaksson & Erlandsson, 1995). After homogenisation and mixing the soil samples were put into 60 ml plastic containers for measurement with a well-calibrated HPGe detector (efficiency 38 % and resolution 2 keV @ 1332 keV), surrounded by a 10 cm thick lead shield, for 48 hours.

Dose rate measurements were performed at all reference sites with the detector RNI 10/SR, placed 1 m above ground, pointing downwards. The measuring time was 5 minutes and an average value of the dose rate was calculated. In addition to the measurements at the reference sites, 20 dose rate measurements were performed at different locations within the city of Göteborg.

TLD-measurements

A group of 25 persons living in the Göteborg area was equipped with 2 TLD (LiF) each, that were carried close to the body 24 hours a day for a period of eight weeks. The persons were instructed not to spend any time outside the area, as defined above as western Sweden, during the measurement and they were also not allowed to fly with airplane. Those who, for some reason, did not wear the detectors were excluded from the study (6 persons). All TL-dosimeters were calibrated before, as well as, after the measurements.

Results and discussion

Reference sites

The results from the reference sites in western Sweden are shown in Table 1. Due to technical problems with questionable data from three of the sites, only 22 of the 25 sites are shown in the table. A comparison between soil samples and field gamma measurements shows a good agreement for the naturally occurring radionuclides, which are supposed to be relatively homogeneously distributed in the ground. For ^{137}Cs the *in situ* measurements show an equivalent surface activity that is generally a factor of 3-4 smaller than the

Table 1. Results from the measurements at the reference communities. The activity for ^{137}Cs calculated from the *in situ* measurements is reported as equivalent surface deposition (not corrected for decay) and for the naturally occurring radionuclides a homogeneous distribution is assumed. Also shown are the ratios between soil and *in situ* measurements.

Site	^{137}Cs Soil	^{137}Cs In situ	^{137}Cs Ratio	^{40}K Soil	^{40}K In situ	^{40}K Ratio	^{208}Tl Soil	^{208}Tl In situ	^{208}Tl Ratio	^{214}Bi Soil	^{214}Bi In situ	^{214}Bi Ratio	Dose rate
	Bq/m^2	Bq/m^2		Bq/kg	Bq/kg		Bq/kg	Bq/kg		Bq/kg	Bq/kg		µSv/h
Skara	959	232	4,1	634	643	0,99	8,5	8,9	0,96	24,0	22,8	1,05	0,13
Uddevalla	1027	342	3,0	559	632	0,88	8,0	8,0	0,99	23,6	25,2	0,94	0,09
Trollhättan	938	250	3,8	747	839	0,89	8,7	10,0	0,87	22,3	24,8	0,90	0,14
Vara	914	313	2,9	648	724	0,89	6,1	7,8	0,78	17,2	25,8	0,67	0,11
Ljungskile	957	235	4,1	632	614	1,03	7,1	7,2	0,98	25,4	28,6	0,89	0,12
Lilla Edet	1635	512	3,2	627	749	0,84	9,5	9,4	1,01	30,2	33,2	0,91	0,12
Stenungsund	1029	268	3,8	613	644	0,95	7,5	9,3	0,81	29,8	34,9	0,85	0,11
Lödöse	1288	314	4,1	566	653	0,87	6,7	8,3	0,81	32,8	41,8	0,78	0,11
Älvängen	1265	398	3,2	713	782	0,91	7,1	8,9	0,80	27,6	33,8	0,82	0,16
Vårgårda	994	305	3,3	566	633	0,89	8,3	9,7	0,86	32,3	37,3	0,87	0,10
Alingsås	897	233	3,8	764	836	0,91	7,2	8,9	0,81	27,3	31,6	0,86	0,15
Kungälv	983	315	3,1	798	874	0,91	8,9	10,2	0,87	27,6	31,7	0,87	0,16
Ulricehamn	1593	366	4,4	800	822	0,97	9,3	11,6	0,80	23,0	28,5	0,81	0,16
Lerum	990	214	4,6	688	702	0,98	6,5	6,8	0,95	30,5	33,2	0,92	0,13
Borås	1447	399	3,6	702	764	0,92	7,1	8,4	0,85	30,4	33,1	0,92	0,14
Hindås	842	265	3,2	542	584	0,93	7,8	8,5	0,92	24,5	26,9	0,91	0,09
Landvetter	890	312	2,9	565	593	0,95	6,4	8,1	0,78	21,4	25,9	0,83	0,10
Mölnlycke	966	276	3,5	649	765	0,85	6,7	8,1	0,83	20,1	25,8	0,78	0,12
Kinna	717	243	2,9	646	691	0,93	6,1	7,3	0,84	26,8	28,9	0,93	0,13
Kungsbacka	1450	414	3,5	791	833	0,95	9,0	9,2	0,97	24,0	27,8	0,86	0,15
Svenljunga	1134	413	2,7	721	778	0,93	7,3	8,2	0,89	25,0	28,5	0,88	0,14
Varberg	1405	376	3,7	608	714	0,85	8,8	9,2	0,96	27,5	30,1	0,91	0,12

activity per unit area calculated from the soil samples. This is, however, a discrepancy of reasonable size, considering the migration of ^{137}Cs down into the soil (Isaksson *et al*., 2000).

The dose rate measurements at the 22 reference communities vary between 0.09 and 0.16 µSv/h and the dose rates seem to be mostly correlated with the activity of the naturally occurring radionuclides, especially ^{40}K ($r = 0.94$). The correlation with ^{137}Cs is weaker ($r = 0.33$).

Results from the measurements at the 6 reference sites at Göteborg are shown in Figure 1. The ratios between soil sample activities and activities determined *in situ* are similar to those found in the larger examination area. The variation in ^{137}Cs activity within the town area may partly be due to soil disturbances since the Chernobyl accident 1986, but may also (at some places)

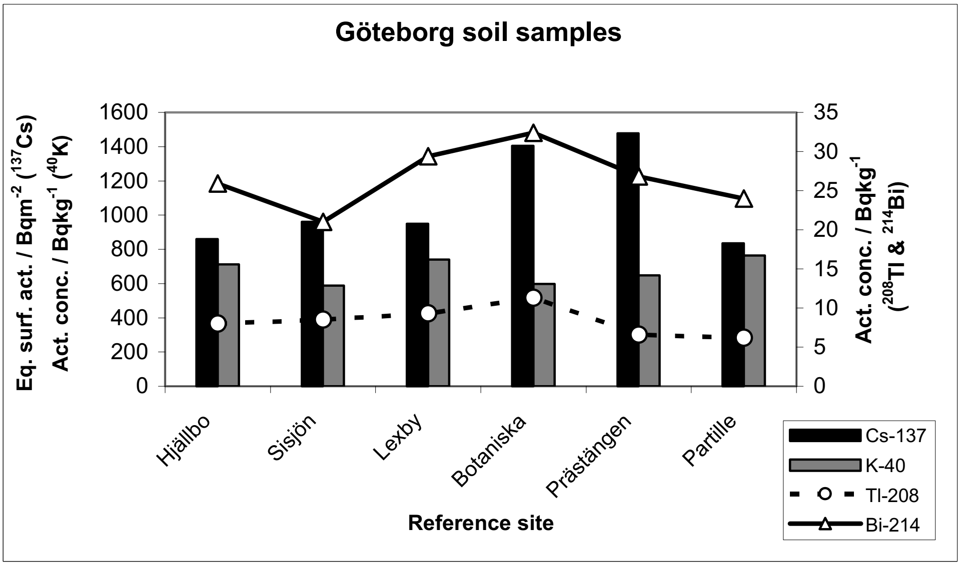

Figure 1. Activity in soil samples from the reference sites in Göteborg.

be a result of the city topography, which influences the amount of precipitation and thereby the deposition. At these sites measurements have previously been performed and the activities of the naturally occurring radionuclides are rather constant, but the activity of ^{137}Cs fluctuates and shows no obvious trend. The reason for this may be that the sites have been disturbed between the measurements and therefore it is assumed that the present measurements serve as a best estimate of the background activity.

The twenty selected sites for dose rate measurements in Göteborg cover a large part of the town and the dose rates vary between 0.09 and 0.16 μSv/h, with a mean of 0.12 ± 0.02 (1 σ) μSv/h. This variation in dose rate at different places in the city is due to both the composition of the ground and the type of ground cover (e.g. asphalt, cobble stones, concrete). The dose rate could thus easily vary by almost a factor of two over a relatively small area and this should be taken into account when selecting reference sites for repeated dose rate measurements.

TLD-measurements

Of the 25 TL-dosimeters handed out 19 were worn during the whole time period (8 weeks) and the mean value for the 19 subjects is 0.11 ± 0.06 (1 σ) μGy/h, which is in good agreement with the dose rates found at the 20 measurement points in Göteborg (mean 0.12 μSv/h).

However, for the 6 TLD:s which were not worn (but left in the person's homes) the mean dose rate is higher: 0.38 ± 0.05 (1 σ) μGy/h. These 6 persons all live in apartment buildings. Similar results have been found in measurements at Gävle (Erlandsson, 2001) and imply that the main part of the dose is received from indoor sources. The ground contamination level may thus have a minor impact on the radiation dose.

Taking this into consideration it may seem strange that the agreement between TLD-measurements and outdoor dose-rate measurements are so good. The TLD-exposure ought to be higher since the persons wearing the TLD:s spend most of their time indoors. However, the radiation environment at the workplace may differ considerably from the radiation environment in the homes where the non-worn TL-dosimeters were exposed. Also, the dosimeters worn by the subjects are shielded to some extent by the body, which could explain the lower signal. The number of TLD:s left at the subject's homes are also too few to draw any firm conclusions and this will be further investigated in a subsequent study planned for 2003.

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Removal of Contaminated Sediments in the Nitelva River

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Abstract

Since 1967 Institute for Energy Technology (IFE), Kjeller, has discharged low-level radioactive wastewater through a special pipeline called the NALFA-pipeline to the nearby river Nitelva. In October 1999 IFE was asked by the Norwegian Radiation Protection Authority (NRPA) to work out a plan for removing radioactive contaminated sediments from the end of this pipeline. A plan was presented in February 2000 and the contaminated sediments were removed during the spring 2000. A total of approximately 180 m^3 of sediments were removed from the river bottom, 16 m^3 from a 10 m^2 area around the end of the pipeline, and the rest from an area of approximately 250 m^2 mainly downstream. The 16 m^3 of sediments from the "hot spot" were filled in 125 drums. Except for 5 drums that are stored at IFE to be used in future research activities, these drums have been deposited in the KLDRA-Himdalen repository in Norway. The remaining 160 – 170 m^3 of sediments were stored in 37 containers and were finally filled in the empty hole after the radioactive waste in the old near-surface repository at IFEs premises at Kjeller had been retrieved.

Introduction

A pipeline called the NALFA-pipeline has been used since 1967 for discharges of low-level radioactive wastewater from IFE to the nearby river Nitelva. The discharges have always been below the discharge limits specified by the Norwegian Radiation Protection Authority (NRPA).

Decommissioning of the Uranium Reprocessing Pilot Plant at IFE-Kjeller in the period 1967 – 1970 resulted in an increased discharge of plutonium and other fission products but still below the discharge limits. Samples of sediments taken in 1971 from the area around the end of the pipeline showed that this area was contaminated. Plutonium, which is known to bind effectively to sediments, was apparent in the samples.

As a part of a program for radiological survey of contamination in the Nitelva, samples of sediments from the end of the pipeline are analysed every year and the results reported to the authorities. Because of the sedimentation process in the river, the area contaminated by plutonium and other fission products from discharges in 1967 – 1970 has been covered. Profiles of sediments taken in 1996 and 1999 showed that the main plutonium concentration was located in a layer of 30 – 50 cm below the river bottom.

The water level of the river and the lake Øyern is lowered every year to meet the flood situation caused by snow melting. This caused the contaminated river bottom to be exposed to air and thus be accessible to the public for shorter periods. In autumn 1999 IFE was requested to remove the contaminated sediments and NRPA asked IFE to work out a report on the extent and level of contamination and a plan for removal. This report was presented in February 2000. Based on this information the NRPA concluded that contaminated sediments containing ^{239}Pu + ^{240}Pu + ^{241}Am above 10 Bq/g (dry weight) should be removed and treated as radioactive waste.

Mapping of the contaminated area and levels of contamination

Mapping of the area of contaminated sediments was carried out in the winter 1999/ 2000 and in March 2000 while the river was still covered by ice. Holes were drilled though the ice and profiles of sediments were sampled down to approximately 60 cm below the river bottom where this was possible (Figure 1). Totally 74 profiles were sampled in the area covering 50 m x 20 m where the end of the pipeline was supposed to be buried in the sediments. The profile samples were sliced in 6 cm sections resulting in more than 500 samples. The content of ^{241}Am in these samples was analysed by g-spectrometry at IFE-Kjeller. Based on earlier measurements of plutonium in these sediments a ratio of 26 between plutonium and americium (dry weight) was used to calculate the concentration of Pu. The measurements and calculations showed that the most contaminated sediments were located between 30 cm and 60 cm below the river bottom. The calculations resulted in a map showing the area where sediments had a concentration of ^{239}Pu + ^{240}Pu + ^{241}Am above 10 Bq/g and therefore had to be removed. Because of the uncertainty in the measurements and in the Pu/Am ratio it was decided to remove sediments having a contamination level of 7 Bq/g and above. The area where the sediments were to be removed is shown in Figure 3. The end of the pipeline is marked with a star. The area represented approximately 260 m^2 and was marked by sticks through the ice.

Removal of contaminated sediments

A private company specialized in underwater technology was hired to perform the work of removing the sediments from the river bottom. Before the work could start, the ice covering the area was removed and a barrier against flooding of the area was established (Figure 2). The water level in the river was slightly lower than the area of the contaminated sediments when the excavation started. Excavation of the sediments was performed with a grab mounted on a mobile crane with an arm that could reach 50 meters outwards (Figure 4 and Figure 5). The crane was parked on the riverbank close to a bridge over the river (Figure 3 and Figure 4). The area along the riverbank, the parking area of the mobile crane and the work area for receiving and pretreatment of sediments were classified as a controlled area and fenced off in order to prevent access by unauthorized persons.

The excavation work started 29 March 2000 and was finished at 3 April 2000. Approximately 16 m^3 of sediments from 10 m^2 around the end of the NALFA-pipeline were loaded into 125 drums and transported to IFE-Kjeller for further treatment (Figure 6). The end of the NALFA-pipeline was located, and the part of the pipeline inside the excavation area was removed. Sediments from 250 m^2 outside this "hot spot" were removed down to about 70 cm and loaded into containers. During the excavation samples were taken and analyzed to ensure that the contamination level in the riverbed was below 10 Bq/g in the excavated areas. All together 160 – 170 m^3 of sediments outside the "hot spot" were loaded into 37 containers and transported to IFE. The local authorities had specified in advance which route to use for transportation out of the area.

During the last days of the work it started to snow and the water level in the river increased and flooded the area thus making the excavation task rather difficult.

Radiation protection measures and doses

To prevent contamination of the riverbank where the sediments were brought ashore and the work area for loading drums and containers these areas were given a thick plastic cover.

Personnel working in close contact with the sediments were required to use special working cloths and were checked for contamination after work. Work inside the controlled area required use of personal dosimeters from IFE's radiation protection service. Doses to IFE-personnel in March and April 2000 showed no increase above a normal level of 0 – 0.5 mSv/month. Doses to hired personnel were below the detection limit of 0.1 mSv/month for the TL-dosimeters. Dose rate measurements made during the work showed a level of 0.1 – 0.15 µSv/h, the natural background level in the area.

Personnel working in close contact with the contaminated sediments were required to use personal air monitors (PAS). The filters in the PAS' were not changed during the preparation and excavation period and were measured with a contamination monitor every day. After the excavation work was finished the filters were measured by a NaI-detector. It was not possible to detect any α-, β- or γ-radioactivity on these filters. The nine filters from the PAS' were also analysed in search for plutonium. Only one of these analysis was successful. The committed dose that could be calculated from the Pu-activity on this filter was 0.2 µSv.

Two air monitor stations were in use 12 hours a day in the period 27 March to 3 April 2000. The two monitors were located at the riverbank facing the contaminated area. The monitors pumped air thought the filters at a rate of 20 l/minute. The filters were not changed during the period. Radioactivity on these filters was measured every day by using a contamination monitor and was used to decide if use of breathing masks was necessary. These measurements and a final measurement of these filters by a NaI-detector showed no traces of γ-radioactivity. The filters were also analysed by a radiochemical method to detect plutonium. The results of concentration of plutonium in air from these measurements were 3.9×10^{-8} Bq/l and 2.6×10^{-8} Bq/l. The background level of plutonium in air was later measured at IFE to be 2.5×10^{-8} Bq/l. From the average value measured by the two monitors on the riverbank and corrected by the background level an average committed dose of 0.04 µSv could be calculated from the work of removing the contaminated sediments.

Waste treatment of contaminated sediments

Sediments in the 125 drums from the 10 m^2 at the end of the pipeline were mixed with concrete. Five drums were stored at IFE-Kjeller to be used in future research activities. The rest of the drums have been disposed of in the KLDRA- Himdalen facility.

The 37 containers with sediments from the 250 m^2 area outside the "hot spot" were stored at IFE for a while. By applying a systematic program for sampling and analysis of the content in these containers it could be proved that the average concentration of ^{239}Pu + ^{240}Pu + ^{241}Am was below 10 Bq/g in all of them. Permission was therefore given by the NRPA to fill these sediments into the old empty near-surface repository at IFE-Kjeller after the radioactive waste was retrieved. This was done in early November 2001.

Figure 1. Mapping of the contaminated area

Figure 2. Barrier against flooding of the area

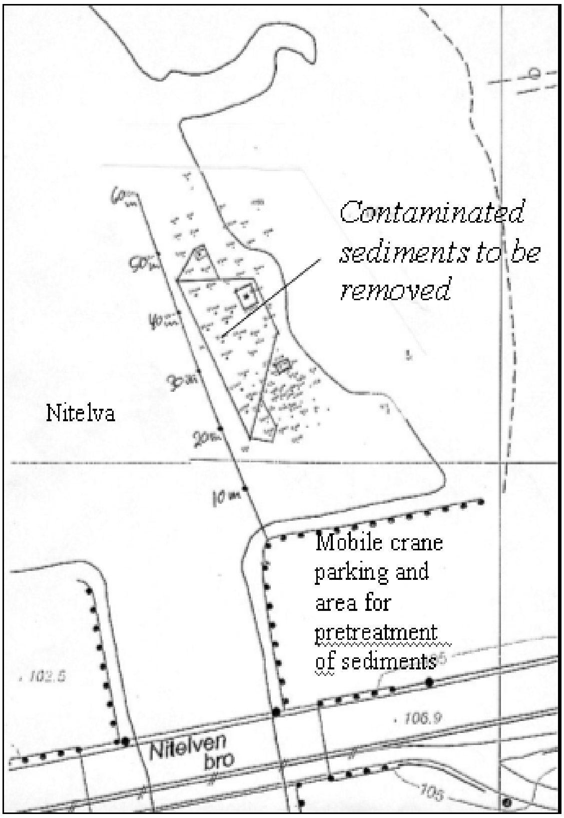

Figure 3. The area with contamination Above 10 Bq/g.

Figure 4. The mobile crane .

Figure 5. Excavation with a grab.

Figure 6. Loading of sediments into drums.

Expanded sediment sampling from Institute for energy technology's waste discharge point, downstream the river Nitelva and into the lake Øyeren. Preliminary report.

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Abstract

"Knowledge is the small part of ignorance that we arrange and classify" Ambroce Bierce.

After a long time of discharging low-activity waste from the Institute of Energy Technology (IFE) into the waters system, a broad sediment sampling research program was initiated to investigate the distribution of radioactive material in the recipient. The research was formed as a project and named ALT-ØYERN. It started in 1998. The intention was to plan and carry out a comprehensive sediment sampling program, including the complete Nitelva/Øyeren water system. A great deal of samples is now collected. The project has adapted and tried out new equipment for sampling. The sampling procedure makes use of plastic test tubes to make sediment cores. The cores are divided into segments making it possible to determine activity depth profiles. The great amount of samples will undoubtedly increase the competence of the section in sampling, sample preparing and radiochemical analysis. The project monitoring will be of great importance as a background status, if IFE in the future wants to perform consequence analysis of its cleaning up of radioactive materials around its former discharge point. As the project still is in the beginning, this paper is to be considered as a preliminary status report.

Introduction

IFE, Institute for Energy Technology has since the early sixties controlled their amount of discharges of radioactive materials and the radiological status in its recipient. The results are reported annually in special reports to the NRPA, Norwegian radiation protection agency. A project named ALT-ØYEREN was started in 1998. A comprehensive sampling of sediments in the river from IFEs former discharge point and downstream to lake of Øyeren was planned. The main intention was to gain adequate information about the dispersing of any anthropogenic nuclides in the water system. At present moment a systematic sampling of sediments from the water system, additional to the regular control samples, is executed. In the sampling procedure new sampling equipment was tested. The equipment makes possible the sampling of sediment cores, which can be divided in segments to observe radioactive depth profiles. Additional reason to run the project is to secure useful information about the circumstances in the recipient, making a background status, before the institute started to clean up radioactive materials around the former discharge point.

This paper also brings forward a tabular form of earlier sediment samples from the river, which might indicate a migration of plutonium. The project will hopefully be able to give more information and knowledge about this.

To sum up, the samples are collected in to:

- Investigate and determine any dispersal of radioactive materials in the water system
- Investigate and determine any migration of radioactive materials from the former discharge point downstream the water system
- Escalate the quality of our analyses program and the all over competence of the staff working in the radiological section.

Sampling positions

The sampling positions are located with a GPS- global position system, shown in figure 1. The GPS position finder is shown in figure 2.

Sampling

Three samples collection were carried out in September 1998. See figure 1.

- The fourth of April samples from Bingsfoss were collected for background measurements
- The eighth of September samples were collected over a distance of 3 km from Nitelven Bridge downwards to a delta called Svellet. Samples nr 21 to 26 was collected over a distance of 750 m between Lillestrøm Bridge and Rælingen Bridge. Samples 31 to 35 were collected over a distance of 3 km in the lower part near the outlet of Nitelva. Sample from point 36 was collected in the middle of the delta Svellet, 2 km from sample point 35.
- The fifteenth of September samples were collected from Teigsåa further down in the watersystem .

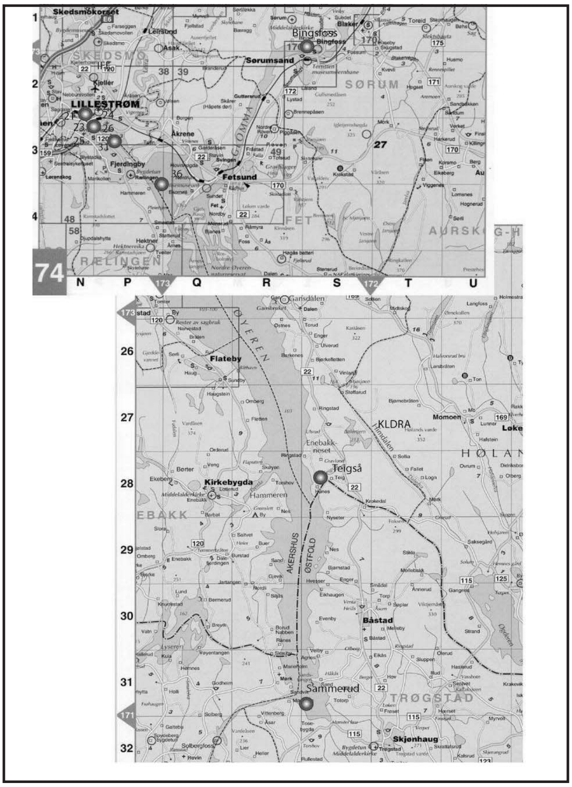

Figure 1. Sampling points.

Teigsåa, 24 km from discharging point, is a small affluent running out into Øyeren close to Sæter. It is well suited for sampling.

Sammerud, representing the most southerly sampling point in the watersystem, is situated 37 km from discharge point.

Bingsfoss is a background point in Glomma, 12 km from IFEs discharge point.

Equipment, procedure

Special sediment sampler providing sediment cores was used for sampling. See figure 2.

A 2,5 cm diameter plastic tube is mounted in a metal holder. The holder is pressed into the sediment and a core sample is produced in the plastic tube. The plastic tube is demounted, closed in both ends to preserve the sample. The plastic tubes are stored in a refrigerator. Two or three samples were collected from the regular sample points. Four samples were collected from the background point. The length of the cores is between 20-35 cm conditioned by

Figure 2. Sampling equipment.

the bottom status. The equipment can be used down to 1,5 m depth. Sampling from greater depth has to be done by a diver.

Samples treatment in the laboratory

In the laboratory the samples are kept in a refrigerator. The sample is partly melted to suitable consistence before dividing into segments. The upper 10 cm of the core is divided into 2 cm thick segments, 0-2, 2-4, 4-6, 6-8, 8-10 cm. The lower part is divided in 5 cm segments, 10-15, 15-20, 20 -25, 25-30 cm. The segments are dried to stable weight at 105 ^{0}C and analysed gamma spectrometric. Some samples are going to be analysed for plutonium.

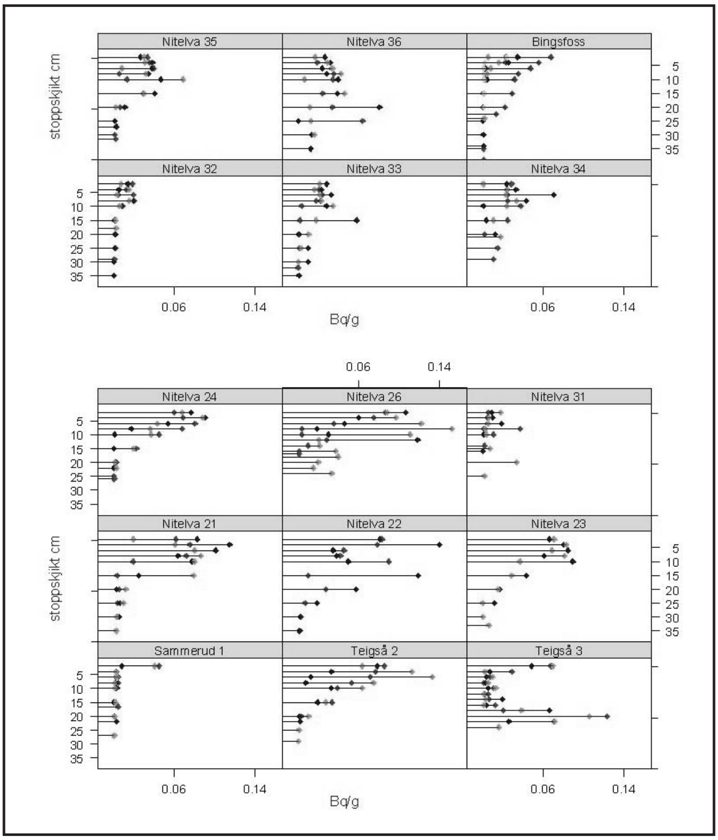

Figure 3. ^{137}Cs results sediment.

Results

The only nuclide showing values above the detection limit is ^{137}Cs. The results of ^{137}Cs are presented graphically in figure 3. Concentrations in Bq/g are shown along the x-axes, and the different depth samples are given in the y-axes. The figure shows some variations in parallel samples from the same samplings point.

Pu-migration

As IFE over a long period of time has controlled and reported the discharging of waste to the Nor-wegian authorities, a considerable amount of sediment sample results exist. The following table shows the values of plutonium from five sample points in the water system, reported since 1985 to 1998. The sampling points are shown in figure 4. The discharge point is situated ca 30 m above sampling point Nybrua.

Year	Kjellerholen SD 1	Nybrua SD 4	Rælingsbrua SD 5	Svellet SD 7	Øyeren SD 8
1985	0,60	15	9	1,3	0,22
1986	0,30	0,6	26	0,6	0,6
1987	0,8	2,9	2,5	0,9	0,3
1988	0,30	16	5,4	2	0,06
1989	0,70	8,1	14	1,7	0,07
1990	0,38	4,3	1,4	4,7	0,8
1991	0,29	7,5	0,56	0,58	0,015
1992	0,33	17	2,6	0,28	0,1
1993	0,48	5,4	4,1	0,5	0,083
1994	0,31	31	7,1	0,49	0,025
1995	0,095	5,2	3,6	0,49	0,035
1996	0,092	2,8	8,7	0,64	0,059
1997	0,2	4	3,4	0,34	0,019
1998	0,014	3,3	4,3	0,32	0,035
Average	**0,34**	**8,8**	**6,6**	**1,1**	**0,17**

The table seems to reveal a migration of plutonium. To enfeeble or to confirm a migration, the project will perform plutonium analyses on a selected part of samples.

From the start of IFEs waste treatment plant and till all discharge of alpha-activity was stopped in 1971, a total of 5 g plutonium is reported discharged. Later calculations related to the cleaning up procedure in year 2000, was reported to be 0,3-0,5 g plutonium in the most contaminated 6 m^2 area, 1 m depth. The calculations are based on exact segments analyses connected to the Ecopraque project (ref 1). The contaminated area is now free-classed. The difference between the total plutonium discharges, 5 g in 1971 and the reported total value in connection with the cleaning up analyses, 0,3-0,5 g and the decreasing values shown in the table, may indicate that some plutonium has migrated down the water system. The project is aware that the two calculations incommensurable.

Summary

The project was started up by enthusiasm and commitment of some research workers who regarded it as an important radio ecological study. The project will not only gain radio ecological knowledge of the water system, but will also by the great number of samples to handle, increase the quality of pre-treatment and analyses in the section. The driving forces are enthusiasm, interest and curiosity. New equipment constructed to sample sediment cores is tested, and many samples are collected. 406 segment samples are analysed. The great number of samples will additionally increase the overall competence of the section in sampling, preparing, analysing, categorizing and reporting.

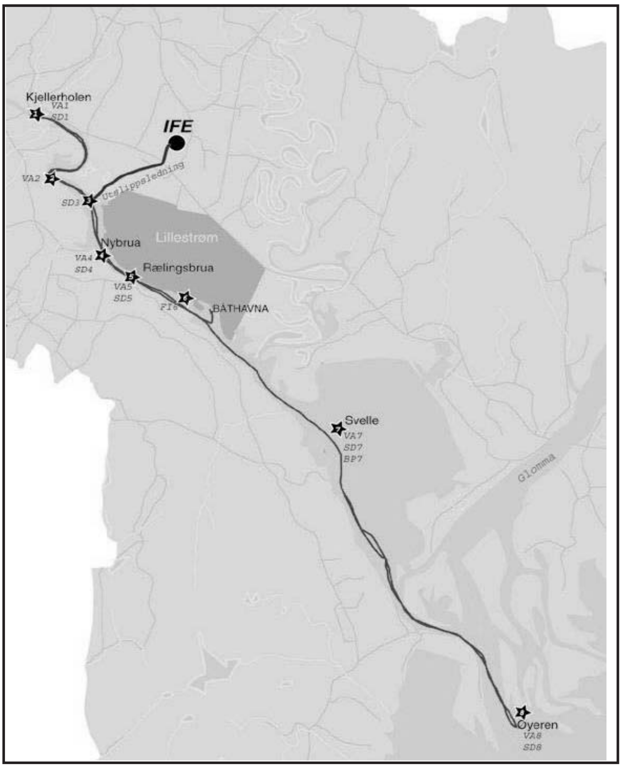

Figure 4. Sampling points in Nitelva.

Scientifically the project will bring forward a greater knowledge of dissemination of anthropogenic radioactive materials in the water system, before and after IFE cleaned up the contamination of radioactive materials around its former discharge point. This knowledge will be of great importance if IFE later on will perform consequences analyses in the water system. All project work is done as teamwork. The project is not given high priority and is not given a budget. It is only performed as time makes it possible. As the project still is in beginning, this paper is to be considered as a status report. Consequently, conclusions at this stadium cannot be drawn. It will hopefully in the in the future end up in a final report. Just before this conference started, the project collected new samples from all sampling points.

Reference 1. Final report ECOPRAQE project, R.N.J Comans (Editor). EC-C-99-104

Radioecology and management of contaminated forests

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Introduction

Dose pathways from contaminated forests to man are related to stay in forests and use of forest products. Their contributions to radiation dose vary according to time after deposition and by groups of people. Wild foods are the major source of long-term internal radiation in areas where people collect mushrooms, berries and herbs, or are hunting for their own use, amounting to at least a few kilograms *per capita* and year. External radiation exposure will hardly exceed internal, even for forest workers. In early phase of fallout situation intercepted radioactive material will retain in tree canopies for a relatively short time. In dense young forests where trees are not too high external exposure is then at maximum because the radiation source in canopy is so close to people in forests and also due to appearance of short-lived radionuclides.

Non-wood forest products are of concern to the public for a long time, but slow contamination of timber can in serious fallout situations cause the most harm to forestry. More than 80% of the income of multiple-use forestry in Finland comes from wood. The availability of domestic timber is a major issue and, if disturbed, also socio-economic consequences of contamination would become substantial.

Multiple-use forestry, in addition to wood production comprises hunting, picking of berries, mushrooms and medical herbs, decorative materials, Christmas tree market, hiking and other recreational use of forests. Only few of these activities as such mean substantial addition to human radiation doses. Regulations for activity concentrations in commodities, also in various forest products in international market, are probably to be applied in the near future as a result of an IAEA project. Long-term internal exposure through wild foods is of most concern to local citizens from the point of view of radiation protection. The multiple-use forestry in northern Europe includes increasing sectors of recreational services and activities for the public. Credible and well-informed intervention is important for all those involved in forestry and using forests.

The aim of this paper is to describe the evidence for effectiveness in reduction of ^{137}Cs uptake, and to appraise practicability of some forest management methods having potential for remediation of contaminated forests.

Intervention

The mitigation of radiation problems in forestry should be in accordance with the general principles of intervention: justification, protection of individual, and optimisation. They are essential and if followed, prevent making bad mistakes in implementation of countermeasures. There are cost-effective strategies for lowering of collective dose through restrictions for the use of wild foods. Less emphasis has hitherto been put on production of acceptable timber using methods that reduce contamination of wood and other forest products. Through efficient surveillance of the contamination the timber resources not contaminated above the acceptable level can be used timely. Remedial actions are needed when adjusted timing of harvesting timber is not sufficient.

Goals of radiation protection concerning contaminated forests are parallel to the goals for assuring access to forests and availability of forest products. However, the stakeholders representing forestry can differ in priorisation of actions from the views of radiation experts, due to their different standpoints. The changes in emphasis of environmental and other 'non-radiation' values among stakeholders are obvious and reflected by forest law in most countries having significant forest industry, in all Fennoscandian countries among others (Hubbard et al., 2002). Principle of sustainable forestry, certification of the whole production chain of timber, increase and development of activities in the field of multiple-use forestry will add to the attributes that are important in intervention. Conservation of nature, protection of forest ecosystem, and consideration of interests of people using and owing forests will set frames for remedial methods that are accounted as acceptable.

The tools for showing the need for intervention are above all measurements of radioactivity, particularly in forests, and predictive dynamic models provided with a data base relevant for the site and stand conditions. Model predictions give guidance for timing of remedial measures, wheras site-specific measurements and known growth conditions direct the choice of intensity of remedial operations. The stand condition and radiation level refer to urgency of the treatment if normal harvesting times are important to forest owners. Options like "no-need to harvest in the next decades" – no action, can also be realistic, depending on later strategies of the forest owner.

The management strategy prepared and accepted before contamination gives a relevant starting point for intervention. Such plans provide value also to on-site adaptation. The same holds true for national forest intervention policy, regional goals and locally.

In the north European conditions the improvement of nutrient status of forests through increased availability of mineral nutrients in soil can significantly reduce the contamination of trees (Table 1). Such methods, namely fertilisation, liming, soil preparation, and prescribed burning are also accepted according to the criteria of sustainable forestry.

Remedial methods for forestry

Field experiments, initiated as early as in the 1960's, have shown that various soil management methods have a significant mitigating influence on radiocaesium contamination of forest vegetation. Applying these methods lowers radiation doses to people in the whole field of forest product utilisation. A series of studies related to soil improvement on various sites and with varying stands are reviewed in Table 1.

Options for soil management

Fertilisation. Fertilisers (N, PK, wood ash) are used to improve site nutrition and nutrient cycling in order to maintain or increase tree growth. Nitrogen is normally growth-limiting nutrient in forested mineral soils and phosphorus and potassium in drained peatlands in Finland. Nitrogen and phosphorus are being applied by using slow-release fertilisers to minimise leaching losses of nutrients and to maximize the fertilisation duration.

Liming. Liming agents have been used to neutralise soil acidity and increase the soil's capacity to buffer changes in pH (Derome et al. 2000). However, the detrimental effects of liming on the growth of Scots pine and Norway spruce in Finland appear to be greater than the benefits of liming (Derome et al. 2000).

Soil preparation. Humus layer is partly broken in order to facilitate artificial or natural regeneration, to decrease the competition of ground vegetation and to increase the temperature of the mineral soil. These effects improve growth conditions and result better regeneration.

Prescribed burning. Understorey vegetation, most of the logging residues and a varying amount of the organic layer are burnt to promote forest regeneration. During the latest decades prescribed burning was replaced by mechanical site preparation because of practical problems, i.e. work dependence on weather conditions, a danger of fire getting out of control and a risk of nutrient leaching. Nowadays prescribed burning is becoming more common in Finland. Prescribed burning releases nutrients from the organic material, improves temperature conditions in the soil and reduces the acidity of the organic layer (Mälkönen et al. 2000). The sites most suitable for this measure are moist and dryish till soils, which do not require draining.

Practicability of soil improvement methods

The effect of a treatment on reduction of caesium uptake by plants can be assessed in advance using data from comparable field experiments and considering the uncertainties in the actual radiation situation. The effects of performed treatments on stands should be surveyed and the gained information used in further planning of intervention.

Table 1. Reducing effect of some soil management methods on ^{137}Cs concentrations in different plant material collected from various mineral and drained peatland soil sites 4 – 12 years after the Chernobyl fallout.

Method	Site	Plant species	Material	Time (years) from application/fallout	Reducing effect, %	Reference
NPK+NPK fert.	Mineral soil	Norway spruce	Wood, bark, branches, needles	18, 13 / 12	73-92	Aro et al. 2002
NPK fert.	Drained peatland	Scots pine	Wood, bark, branches, needles	34 / 9	8-23	Kaunisto et al. 2002a
PK/NPK+NPK+PK fert.	Drained peatland	Scots pine	Wood, bark, branches, needles	38, 18, 1 / 9	33-58	Kaunisto et al. 2002a
PK/NPK fert.	Drained peatland	Scots pine	Needles	25 / 4	47	Kaunisto et al. 2002b
PK/NPK+NPK fert.	Drained peatland	Scots pine	Needles	25, 13 / 4	60	Kaunisto et al. 2002b
NPK+NPK fert.	Mineral soil	Ground vegetation	Above-ground parts	18, 13 / 12	50-85	Moberg et al. 1999
PK+K fert.	Drained peatland	Ground vegetation	Above-ground parts	22, 9 / 12	32-62	Moberg et al. 1999
Ash fertilisation	Mineral soil	Lingonberry	Berries	2 and 7 / 6 and 11	23-78	Levula et al. 2000
Prescribed burning	Mineral soil	Lingonberry	Berries	7 / 11	55	Levula et al. 2000
Liming	Mineral soil	Scots pine	Woody parts Needles	5 / 6 5 / 6	17-28 15-17	Rantavaara & Raitio 2002
Soil preparation	Mineral soil	Scots pine	Woody parts Needles	5 / 6 5 / 6	46-58 57-62	- „ -
Liming + soil preparation	Mineral soil	Scots pine	Woody parts Needles	5 / 6 5 / 6	58-71 66-69	- „ -

There are restrictions for forestry operations arising from environmental and landscape protection. For instance, large clear cutting areas should be avoided because of landscape values, biodiversity, susceptibility to erosion and increased risk of contamination of shallow ground waters. Also protection of the ecosystem is taken care of, when soil preparation is not deep ploughing but rather harrowing, scarifying or mounding.

Demand for protection of biodiversity is delicate in the short term for instance when fertilisation or soil amendment with wood ash changes the structure of species in field vegetation. Later such changes often gradually disappear and the initial state returns to the site. Big mammals prefer grass feed if available in summer, and thereby fertilisation reduces also contamination of game meat. For example, elk does come to fertilised sites, abundant in grass, for feed. The microbial activity in humus will probably change after amendment of soil using wood ash (Perkiömäki & Fritze, 2002), but is it harmful or not, has not been clarified sufficiently. Methods based on good forest management practices would mostly be acceptable, if stakeholders are well informed of the plans, expected effects, implementation, costs, and follow-up of intervention.

The costs for remedial measures are typically estimated comprising working time, material and equipment costs, and possibly secondary costs for wastes. However, wastes are not generated in the operations referred to. The profitability of fertilisation as a remedial method can often be shown, which is a noteworthy factor when the scale of intervention is designed.

Technical feasibility of forest management methods and capacity for implementating them as countermeasures are based on local use of the actual methods. Eventual capacity problems can be reduced by adjusting the schedule of intervention; some flexibility in timing is possible as changes of radioactivity in timber are slow.

Harmful socio-economic consequences of forest contamination would be reduced through improved functioning of the whole field of forestry. Remediation for acceptable timber will also improve the availability of acceptable wild foods.

Conclusions

Cost-effective, practicable remediation of contaminated forests can be based on customary forest management methods, if designed using relevant multidisciplinary expertise. It is possible to make profitable investments and yield timber at normal harvesting times during the rotation period while the local contamination exceeds the level for production of acceptable timber without intervention. The actual intervention causes hardly any, or very low, direct costs for forestry operations, if the sites are carefully chosen concerning the expected effect of remediation, and related harvesting plan. This should motivate authorities and industry for systematic measurements of radioactivity, which would further improve the results through guidance for timing and adjustment of intensity of remediation. There is potential for successful remediation with methods which simultaneously fulfil the goals of forestry and reduce the uptake of radiocaesium. Forest owners and their consultants need practical advice and options prepared for their remediation planning. An optimised response to a contaminating event can largely prevent any disturbance of round wood market and utilisation of forests by local people.

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Radioactive Contamination of Agricultural Products in Poland

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Introduction

Radiological contamination of the environment is caused by nuclear activities on the globe: nuclear weapon tests and the Chernobyl accident. The transfer of radionuclides to the organism via ingestion is one of the sources of doses obtained by people. To assess the doses received by humans the intake of isotopes with daily diet was defined. The concentration of radionuclides in foodstuffs was determined.

The network of Service for Measurement of Radioactive Contamination systematically controls all kinds of important agricultural products such as milk, meat, vegetables, fruit, cereals and forest products: mushrooms, blueberries etc. Measurement stations involved in food monitoring act within Sanitary-Epidemiological Stations, Veterinary Hygiene Units and Chemical - Agricultural Stations. All activities are co-ordinated by the Central Laboratory for Radiological Protection [1].

The level of activity of caesium isotopes has regularly been monitored in collected samples originating from different administrative districts of Poland. Since 1994 the ^{134}Cs concentration has been below the detection limit. The activity of ^{137}Cs has been measured to determine long-term effect of the accident on the contamination of milk, meat and other foodstuffs.

Materials and methods

The sampling rate has been dependent on the radiological situation. Before the Chernobyl accident food samples were collected on monthly, quarterly or annual basis of all administrative districts in sampling points. Shortly after the Chernobyl accident milk was collected daily, meat, fish, poultry and eggs weekly. The most common kinds of vegetables and fruit were sampled daily, particularly these ones, which were ready for consumption at that time.

Since 1987 a special sampling program has been valid. Milk has been collected weekly and the average monthly sample has been tested. Meat (of different kind), fish and poultry have been collected once per 3 months, eggs once a year, and the most popular vegetables and fruit while harvesting.

The activity of caesium isotopes was determined by radiochemical method or gamma spectrometry. For radiochemical determination the samples, after ashing and dissolving in nitric acid, were filtered through a radiochemical funnel with ammonium molybdophosphate (AMP) bed, selective for caesium. The activity of caesium in the bed was determined using a counter with thin plastic scintilator.

For the spectrometric measurement in Marinelli geometry a gamma spectrometer with HPGe or NaI(Tl) detector was used.

Results and discussion

Before the Chernobyl accident activity of ^{137}Cs in foodstuffs was below 1 Bq kg^{-1}. The regional differences were not observed over the whole territory of Poland. ^{137}Cs in foodstuffs was the consequence of ^{137}Cs accumulated in soil as a follow up of nuclear weapon tests. The situation changed drastically immediately after the Chernobyl accident (April, 1986) when the ^{137}Cs and ^{134}Cs nuclides appeared in large quantities, both in air and total fallout, resulting in the soil, pasture and plants contamination. The pasture season for cows begins early in May in Poland. The distribution of contamination was non-uniform with significant local differences. The north-east area and south part of Poland were the most contaminated area. The low level of contamination was registered in west Poland [2].

Milk is important for the monitoring of radioactive contamination as radionuclides appear in it immediately after a cow has eaten contaminated fodder, as well as the fact that it has great share in our diet. Mean activity of ^{137}Cs and ^{134}Cs in milk was 25 Bq l^{-1} and 13 Bq l^{-1} respectively in May 1986. In samples of milk from some cows the activity of several hundreds Bq l^{-1} was found. In June the contamination level decreased approximately by half, during the next months was dropping gradually. From the beginning of 1987 an increase was observed again. This was caused by feeding cows with hay collected in June 1986 [3].

In the following years the activity of caesium isotopes in milk was decreasing and it reached a level of approx. 1 Bq l^{-1} at the end of 1992. Activity of ^{134}Cs has been below detection limit in most samples of milk since 1992. Figure 1 shows the changes of ^{137}Cs in mean monthly samples of milk in Poland during the period between 1986-2001.

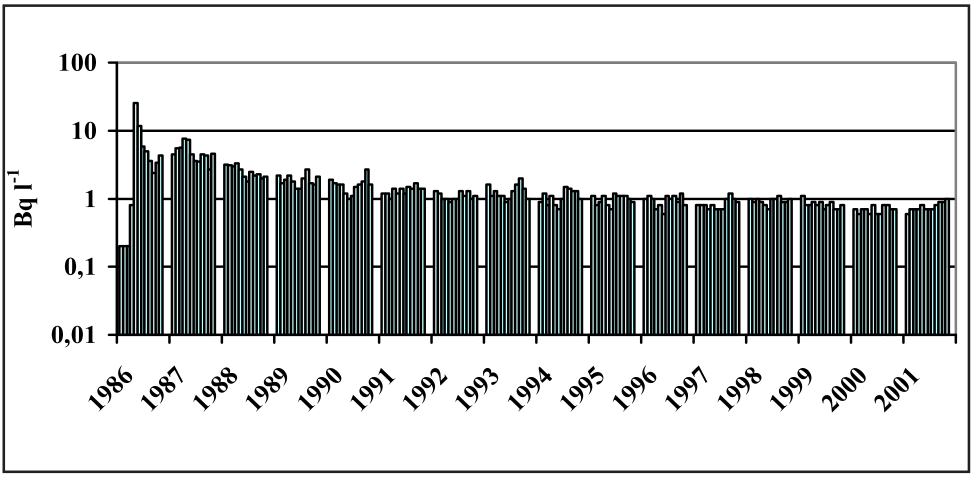

Figure 1. Mean monthly activity of ^{137}Cs in milk in Poland, [Bq l^{-1}].

Four kinds of meat were tested: pork, beef, mutton and horse meat. Until the middle of May 1986 the concentration of caesium isotopes did not exceed 26 Bq kg^{-1} of ^{137}Cs and 14 Bq kg^{-1} of ^{134}Cs but it increased later because of the accumulation of these isotopes in animal organism. In some cases between June and August 1986 the concentration of both caesium isotopes exceeded 500 Bq kg^{-1} – 800 Bq kg^{-1} in single samples. Table 1 shows the activity of caesium isotopes in different kinds of meat in 1986 -1988.

Table 1. Activity of ^{134}Cs + ^{137}Cs in meat in Poland in 1986-1988, [Bq kg^{-1}]

	^{134}Cs + ^{137}Cs Bq kg^{-1}	Beef	Mutton	Horse Meat	Pork
1986	mean range	50 <5 – 527	57 <5 – 860	65 7 – 594	35 6 -316
1987	mean range	45 <5 - 622	53 <5 - 533	61 <5 - 501	21 <5 - 106
1988	mean range	19 <5 - 226	21 <5 - 209	32 <5 - 728	10 <5 - 44

During the few years the level of meat contamination has been decreasing systematically. Since 1994 the activity of ^{134}Cs was below the detection limit in most samples. The average activity of ^{137}Cs has been on the level of 2 - 3 Bq kg^{-1} reaching 50 Bq kg^{-1} in few samples. The differences in activity between kinds of meat were less visible. In 1986 the contamination of poultry and fish was below some tens of Bq kg^{-1} and decreasing gradually. At present the activity of ^{137}Cs in poultry is below 1 Bq kg^{-1}. Mean concentration in fish is on level of the 2 Bq kg^{-1} reaching 20 Bq kg^{-1} in single samples.

The season of vegetation starts in May and during summer month most of fruit and vegetables is harvested. Contamination of the environment after the Chernobyl accident caused high contamination of these plants. The highest average activity of $^{134}Cs+^{137}Cs$ in fruit was registered in currant 200-400 Bq kg^{-1}, in single samples even 2000 Bq kg^{-1}. Fruit picked in September and later was less contaminated. Activity of both caesium isotopes in vegetables and cereals was less significant except leafy vegetables picked in May and June, in which the activity exceeded some hundreds of Bq kg^{-1}. Already in 1987 the content of accumulated radionuclides significantly decreased in comparison to 1986 and since 1989 the average activity has been below 1 Bq kg^{-1} as is shown in table 2.

Table 2. Activity of $^{134}Cs+^{137}Cs$ in fruit and vegetables in Poland, [Bq kg^{-1}]

	Fruit Bq kg^{-1}			Vegetables Bq kg^{-1}		
		mean	range		mean	range
1986	black currant	385	31 –2201	cabbage	13	<5 – 93
	other currant	291	26 – 1769	carrot	17	<5 – 93
	gooseberry	120	13 – 1219	potatoes	17	<5 – 49
	raspberry	79	<5 – 546	other		
	strawberry	28	<5 – 205	vegetables	28	<5 - 215
	apple	31	<5 – 323			
	plum	29	<5 - 141			
1987	currant other fruit	10 3,8	<5 – 44 0,3 – 35,7	all kinds of vegetables	1,2	0,3 –15,4
1988	currant other fruit	 2,0	<5 – 20 0,3 – 9,8	all kinds of vegetables	1,0	0,3 –17,9
1989 - 2001	all kinds of fruit	0,8 – 0,5	<0,1 – 19,2	all kinds of vegetables	0,9 – 0,5	<0,1 –11,4

Summary and conclusions

The Service for Measurement of Radioactive Contamination has assured a continuous monitoring of the foodstuffs on whole territory of Poland. During the period between 1986 –2001 some hundreds of each kind of tested samples were analysed. The great amount of samples has allowed a good statistic dose assessment [4].

Before the Chernobyl accident there were no territorial differences in the radioactive contamination foodstuffs. Fallout originating from nuclear explosions in the world covered Poland uniformly. After the Chernobyl accident the situation changed. There have been regional and even spot differences in contamination. Radioactive contamination of ^{134}Cs according to half-life lost importance in 1994. As the long-term effect on environment the activity of ^{137}Cs has been determined. Since 1994 mean contamination of fruit, vegetables, cereals, eggs, poultry has been on the same level as it was before the accident. Higher radioactivity remains in milk, meat, fish and forest products.

Based on the data concerning activity of caesium isotopes in different kinds of foodstuffs and statistical data of mean intake of these foodstuffs, annual mean intake of ^{134}Cs and ^{137}Cs via ingestion has been calculated . Also an annual effective dose related to the consumption of contaminated food could be assessed for the period between 1985 and 2001. Table 3 presents annual mean intake and the doses of ^{134}Cs and ^{137}Cs obtained by population of Poland. Having considered the diversity of dietary habits of consumers and their place of living, the effective dose received via ingestion may even be five times higher.

In Poland the accepted effective dose caused by contamination of the environment with artificial radionuclides is 1 μSv. The doses obtained by the population were some percent of this value.

Table 3. Annual mean intake of ^{134}Cs and ^{137}Cs [Bq per year] and mean annual effective dose [μSv] due to these radionuclides intake via ingestion in Poland.

	Annual mean intake [Bq per year]		Annual mean effective dose [μSv]	
	^{134}Cs	^{137}Cs	^{134}Cs	^{137}Cs
1985	-	325	-	4
1986	2054	4324	34	54
1987	805	2246	13	28
1988	247	1014	4	13
1989 – 1992	136 – 39	939 – 605	3 – 1	12 – 8
1993 - 2001	<25 - <1	about 500	-	6 -7

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The role of fungal mycelium in the transfer of radionuclides in forest soil

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The activity concentrations of Cs-137 in fruitbodies of fungi have been studied intensively after the Chernobyl accident. The levels of Cs-137 in fungal fruitbodies have been high – in some species very high. The typical range has been 10 to 100 times higher than that found in plants from the same forest area (Olsen 1994). There are several pathways for the transfer to man. By picking mushrooms and consume them the intake of Cs-137 can be significant particularly if we eat some of the high-levels species. Another pathway is via animals – for example game animal (Johanson 1994) which eat large quantities of mushrooms and we eat meat from them. These are the two pathways which often have been discussed but, additionally, fungi is involved in nearly all of the transfer of Cs-137 from soil or litter to the fungal mycelium and the transfer further to nearly all plants of the forest ecosystems. An important role of fungi in the forest soil ecosystem is to decompose organic substanses, to accumulate various mineral nutrients and also caesium and in exchange with photosynthetic products from the plants transfer these mineral nutrients to the plants. In a mineral nutrient poor forest soil nearly all mineral nutrients will pass through the fungal mycelium. In exchange about 10 to 20% of the photosyntetic products produced by the plants will be supplied to the fungi (Smith & Read 1997).

The role of fungal mycelium in the metabolism of Cs-137 in forest soil is ignored by many radioecologists – many of the models trying to explain Cs-137 tansfer from forest soil do not mention fungi at all and also in reviews of Cs-137 and organic substances in soil fungi is often not mentioned at all. Thus the most important parameter is not used in the attempt to explain the binding and accumulation of Cs-137. One of the main problems is to find methods for studying the various transfers – soil-fungi-plant. The biological controls could be found at various parts of this chain of events. First, the uptake of caesium from soil or even directly from litter and minerals to the fungal mycelium can be under biological control. Second, there could be some “excretion” of mineral nutrients from the fungal mycelium, and whether these include caesium is not known. As a third step, we have the transfer or exchange of caesium occurring from the mycorrhizae formation to the plants in exchange of photosynthetic products. There must be a strict biological control of this exchange since the Cs-137 activity concentration in the fungal part is 10 to 100 times higher than in the plant part. One reasonable strategy for the fungi seems to be to take up as much mineral nutrients as possible and transfer as little as possible to the plants.

There are many difficulties in studying the dynamics of the caesium transfer in the fungal systems but it is possible to take some small steps, and we have used two approaches. The first was to use a mechanical fractionation instead of a chemical extraction (Nikolova et al 2000). Soil samples were collected from various forest stands – normal forest soil, sandy soil and peat soil. In the laboratory the soil samples were mechanically separated on 2 mm mesh into three fractions: bulk soil, rhizosphere fraction and a residue which we called the soil-root interface fraction. As a mean value, the percentage of organic matter in the bulk soil fraction was 69%, in the rhizosphere fraction 70% and in the soil-root interface fraction 95%. Of the total Cs-137 activity in the soil, 45% as a mean was found in the bulk soil fraction, 37% was found in the rhizosphere fraction and 18% in the soil-root interface fraction. The highest Cs-137 activity concentration – usually 3 to 4 times higher than in other fractions - was always found in the soil-root interface fraction. The results thus show that 10 to20 % of the Cs-137 inventory of the soil was in some way associated with the biological part of the soil.

Another approach was to isolate the fungal mycelium from soil (Vinichuk & Johanson 2002). Fruitbodies of various species were collected in Ukraine, and soil samples in close vicinity of the fruitbody were also collected and sliced into 1 cm layers. The mycelium was prepared from the soil using tweezers under a microskop (64X magnification). The Cs-137 activity was determined both in soil and in mycelium. The highest Cs-137 activity concentrations were found in the upper part of the soil profile both in the soil and in the mycelium. The Cs-137 activity concentration was usually higher in the fruitbodies compared with the mycelium with a mean concentration ratio of 10. The percentage of the total Cs-137 inventory in the soil found in the fungal mycelium ranged from 0.1 to 50% with a mean value of 15%. Both studies showed that a rather large part of the Cs-137 in the soil – 10 to 20% - was already within or at least associated with the fungal mycelium. In the above ground part of the vegetation in a forest between 5 and 10 % is usually found.

Thus the fungal mycelium seems to act as a sink of Cs-137, as suggested by Olsen et al (1990), and seems to have the capacity to accumulate most of the plant available Cs-137 in the soil. It is known that mycorrhizae fungi also can decompose organic matter although the saprophytes (decomposers) seem to be more effcient to performe that. There might be a direct transfer of Cs-137 from litter to mycelium when the mycorrhizae fungi decompose organic matter, but it has also been suggested that there is a similar direct transfer from the saprophytic fungi to the mycorrhizae fungi (Lindahl et al. 2002). If so, there are pathways for Cs-137 from litter to saprophytes and further to mycorrizae and plants without passing the soil solution.

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Posters

11a Nuclear fuel particles in the environment - characteristics, atmospheric transport and skin doses

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In the present thesis nuclear fuel particles are studied from the perspective of their characteristics, atmospheric transport and possible skin doses. These so called hot particles can be released into the environment, as has happened in the past years because of human activities and different incidents and accidents, for example in a severe nuclear accident causing an acute health threat. Nuclear fuel particles with diameter of tens of micrometers, referred here to as large particles, may be hundreds of kilobecquerels in activity and even an individual particle may produce deterministic health effects.

Health threat caused by hot particles differs to that of uniform contamination. Although their number in the air is small and, thus, the average concentration of radionuclides is small, there is a possibility that a particle may come in contact with tissue. A highly active beta-emitting particle may then cause a large but very local dose to the tissue, whereas at distances larger than about one centimetre from the source the dose is negligible. Quantities such as effective dose equivalent are then not necessarily appropriate for characterising health effects caused by hot particles.

Detection of individual nuclear fuel particles in the environment, their isolation for subsequent analysis and their characterisation are not straightforward and requires well-designed sampling and tailored analysis methods. In the present study the need of the development of particle analysis methods is highlighted. It is shown that complementary analysis techniques are necessary for proper characterisation of the particles. Methods routinely used for homogeneous samples may produce erroneous results if they are carelessly applied for the particles. Consequently, they may misrepresent the threat caused by hot particles.

Large nuclear fuel particles are transported in the atmosphere differently compared with small particles or gaseous species. Thus, the trajectories of gaseous species are not necessarily adequate for calculating the areas that may receive fallout of the large particles. A simplified model and a more advanced model based on real-time weather information were applied in the case of the Chernobyl accident for calculating the transport of the particles of different sizes. The models were appropriate in characterising general properties of the transport but were not able to properly predict the transport of the particles with aerodynamic diameter tens of micrometers, detected in the environment hundreds of kilometres from the source, using only the present knowledge. Either the effective release height has been higher than reported previously or convective updraft may have influenced on the transport. Models applicable to large particles dispersion in a turbulent atmosphere should be further developed.

Large particles are poorly inhalable because of their size. They may be deposited in the upper airways but are not easily transported deep into the lungs. Instead, deposition onto the surface of skin is relevant with respect to acute deterministic effects. In the present work skin doses are calculated for particles of different sizes and different types by assuming that particles are deposited on the body surface. The deposition probability as a function of the number concentration of the particles in air is not estimated.

The doses are calculated at the nominal depth of the basal cell layer and averaged over a square centimetre of the skin. Calculated doses are compared to the annual occupational skin dose limit for the public (50 mGy at depth of 0,07 mm and averaged over 1 cm^2). After the Chernobyl accident the most active nuclear fuel particles detected in Europe, even hundreds of kilometres from the source, would have been able to produce a skin dose exceeding this limit within one hour when deposited onto skin. However, the appearance of deterministic effects necessitates more than one day lasting contact to skin.

Health hazards caused by nuclear fuel particles must be taken into account in estimating the consequences of a severe nuclear accident.

11b I-131 in air filters at Gävle

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Abstract

The sampling station for both radioactive air- and deposition samples at Gävle, Sweden, was reopened in December 2000. During the 13 months from January 2001 to March 2002, ^{131}I has been detected at eight occasions. There is only one other station in Sweden (Visby), which has reported ^{131}I but on one occasion only. The source of radiation has to be local. The air sampling station is situated at the sewage treatment plant Duvbacken where samples of sewage sludge have been collected and the activity concentration of ^{131}I and several other radioactive nuclides have been measured. At the same time meteorological observations of wind direction and speed have been made not far from the sewage treatment plant. We have found a good correlation between the administration of cancer treatment doses for tyroidea cancer at the general hospital at Gävle, favourable wind direction from the sludge thickeners to the air filter, and the concentration of ^{131}I in air filters and in outgoing sludge from the sewage treatment plant.

Introduction

After the Chernobyl accident in 1986 the network of sampling stations for ground level air in Sweden was extended with 7 stations in the Gävle area. These were in operation until the end of 1990 and at the end of 2000 only 6 stations were working in Sweden.(Vintersved and De Geer, 1982). In connection with the extensive sampling of sewage sludge from the sewage treatment plant Duvbacken at Gävle and as there were indications that much of the ^{7}Be and ^{137}Cs in the sewage sludge was due to resuspension air sampling station was restarted at the end of 2000. Normally it is only the naturally occurring radon daughters, ^{7}Be (due to the cosmic radiation) and ^{137}Cs which is due to the Chernobyl accident that are detected in the air filters. Just after the accident a lot of shortlived nuclei were also detected both in the air filters and in the sewage sludge. Among them ^{131}I with a half life of 8.04 d. But already after about 2 month all traces of this nucleus had vanished.

Materials and methods

The air sampler has a capacity of 10^5 m^3/week (Vintersved and De Geer, 1982). The air sample filters are collected weekly and compressed into small discs with a diameter of 60 mm and thickness of 13 mm. These samples are measured 3 - 4 days after collection with a well shielded HPGe - detector.

Samples of outgoing water and sludge have also been collected at the sewage treatment plant. The water was passed through paper filter, and anion and cation exchangers. The resins were dried and measured in 180 ml tubs with a HPGe - detector and the activity concentrations of ^{7}Be, ^{40}K, ^{131}I, ^{137}Cs, ^{228}Ac, ^{238}U, and ^{226}Ra were determined. (Erlandsson and Mattsson,1978; Ingemansson, *et al* 1981) The ratio between the activities in the sludge and the outgoing water is 2.5 ± 0.5 % for ^{131}I. (Erlandsson and Eklund, 2001).

Meteorological observations of wind speed and direction have been made at a station 5 km from the sewage treatment plant.

Results and discussion

The results from both the air filter and sewage sludge measurements are shown in Table I. Also shown in the table are the date of administration of all treatment doses for tyroidea cancer and the strength of the doses during the period of observation.

Table I. Activity concentration of ^{131}I in air filters, sewage sludge and administered treatment doses.

Date	Air filter Act. Conc. (µBq/m³)	Date	Adm. activity (MBq)	Date	Sew. sludge Act.conc. (Bq/kg)
2001					
15/1-22/1	1.5 ± 0.3	15/1	3700	17/2-23/2	970
5/2-12/2	0.6 ± 0.2	9/2	1200	24/2-2/3	130
12/2-19/2	3.0 ± 0.3	12/2	7400	3/3-9/3	182
24/9-1/10	2.4 ± 0.5	24/9	3700	13/10 –19/10	133
10/12-17/12	0.6 ± 0.3	11/12	3700	22/12-28/12	222
2002					
31/12-7/1	0.6 ± 0.3	9/1	7400	12/1-18/1	284
7/1-14/1	1.7 ± 0.3	9/1	1300	26/1-1/2	230
11/3-18/3	0.8 ± 0.3	11/3	7400	30/3-5/4	113

Most of the ^{131}I which has been given to patients at the General Hospital at Gävle is excreted with the urine within a couple of days. It takes about 24 hours before the ^{131}I reaches the first sludge thickener, where it stays for 12 to 36 h in the sewage treatment plant. The thickeners are basins with a diameter of 15 m and the first is situated only 18 m from the air sampling station. The lay-out of parts of the sewage treatment plant is shown in Fig 1. There are 3 thickeners where the sewage water is treated so that the dry substance is 5-6%. Because the thickeners have such a large area and because the distance is short the air filter is surrounded all the time when ^{131}I is passing the thickeners by an evaporation cloud. Favourable wind directions may enhance the transport of ^{131}I . The air filter is placed almost due east of the first thickener (Fig. 1) and the most favourable wind direction should be between WSW and W, which is covered by the sectors WSW and W which have a high frequency of direction readings. Frequencies of wind directions are shown in Table II. Not only is there a registration of ^{131}I in the air filter every time a big dose of ^{131}I have been administered but also 2 to 3 weeks after, there is a marked peak in the activity concentration curve in the outgoing sludge (Table I).

We have also on some occasions measured ^{131}I at Visby, but we have not been able to find the source.

In an example we will follow the transport of the activity of a treatment dose of 3700 MBq administered 24/9 2001. The airborne ^{131}I- activity leaving the plant can be estimated as follows. In the air filter for the week 24/9 to 1/10 an activity concentration of 2.4 ± 0.5 μBq/m^3 was measured. The air volume

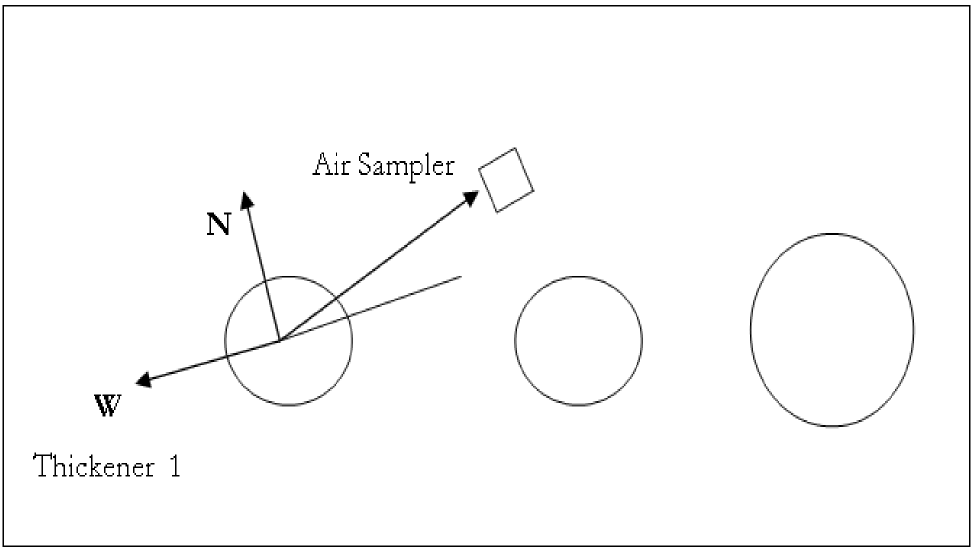

Fig. 1. Orientation of sludge thickeners and air sampler at Duvbacken, Gävle.

Table II. Frequency of wind directions (%).

		NNE+NE	ENE+E	ESE+SE	SSE+S	SSW+SW	WSW+W	WNW+NW	NNW+N
2001	15-22/1			7	11	44	9	2	7
	5-12/2				28	27	2	2	
	12-17/2				9	27	27	25	13
	24/9-1/10	4			9	27	25	14	9
	11-17/12				2	22	15	15	38
2002	31/12-7/1	4			5	43	17	4	5
	7-14/1				4	54	41	3	
	11/18/3	4		4	11	21	32	4	10

that had penetrated the filter was 103 700 m^3. This means that a total of 0.25 was captured on the filter. The glass fibre filter used can be regarded as 100% efficient for particles of all sizes but will not capture iodine in gaseous form. Since we do not know how much of the iodine that leaves the basin in gaseous form we assume that only 10% of the iodine is attached to particles, which was roughly the case for the the activity from the Chernobyl accident. Thus the total activity passing through the filter area of 0.314 m^2 is 2.5 Bq at an approximate distance of 25 m from the source. If the activity from the basin was evenly distributed through a half sphere of radius 25 m the total activity leaving the basin would be $(2.5/0.314) \times 2\pi \times 25^2$ Bq = 31 kBq. Since the sampler is effective for iodine leaving the basin only when the wind direction is SSW, SW, WSW and W we assume that the captured iodine is a sample from an evenly distributed activity into a quarter of the half sphere which gives 8 kBq as the source activity. From the frequency of win directions presented in Table II, one can see that during this week the wind direction was in the four sectors 52% of the time. Thus our 8 kBq should be doubled which means that roughly 16 kBq leaves the thickener via the air. This activity is negligible compared to the activity leaving the plant via the sludge phase 88.7 MBq and via the water phase 3550 MBq

The activity concentration in the sludge is given in Table III, column 2 and the total activity of the sludge in column 3. The maximum activity leaves the plant 2 – 3 weeks after the injection. The activity corrected for the half-life (8.04 d) is given in column 4. Because of the short half-life the correction factor is rather big. The activity is not only due to the big dose but also to smaller doses of 300 to 500 MBq administered about once a week which affect the total activity concentration in the sludge. The summed activity in the sludge is 88.7 MBq and with a sludge to water ratio of 2.5 ± 0.5 % the overall The activity leaving the plant is 88.7 MBq(sludge) + 3550 MBq(water) + 0.016 MBq(air), which gives a total of 3600 ± 750 MBq compared to the administered dose of 3700.

Table III. Activity balance.

Date	Sludge	Sludge (total)	Sludge (time corr.)
2001	(Bq/kg)	(MBq)	(MBq)
3-7/9	9	0.273	
8-14/9	20	0.600	0.60
15-21/9	86	2.606	5.66
22-25/9	127	4.077	14.85
28/9-5/10	133	3.830	25.51
6-12/10	69	1.795	21.30
13-19/10	36	0.634	14.12
20-26/10	10	0.176	7.16
27/10-2/11			Sum 88.7

The agreement is rather good and the activity concentration over the thickener is well below the allowed dose. Although the activity leaving the basin through the air is negligible compared to what is leaving the plant through the sludge and water it is still enough to give a clear signal in the airfilter. This investigation points out the importance of accurate knowledge about the medically used radionuclides detected in sewage sludge.

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11c Airborne Cesium-137 in Finland After the Chernobyl Accident Based on the Measurement of Archived Air Filter Samples

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Introduction

The huge pressure and explosions in the Chernobyl nuclear power plant accident caused a release of artificial radionuclides into the atmosphere. 10 – 20 % of the volatile nuclides, e.g. ^{137}Cs, present in the reactor inventory was distributed into the environment. The energy released in the hydrogen explosion and heat produced by burning graphite and decaying fission products caused the radioactive debris to reach considerable altitudes, where high wind speeds were prevailing. This caused the debris to spread quickly in the atmosphere. The plume was distributed practically all over the northern hemisphere, its advance being monitored both by measurements and by air-mass trajectory calculations.

The Finnish Meteorological Institute has collected weekly aerosol samples for radioactivity monitoring purposes at ten stations (Figure 1) in Finland since the 1960s'.

The samples are measured at the FMI's laboratory e.g. for long-lived beta activity (total beta activity five days after the end of sampling). This long-lived beta activity consists of lead-210 and artificial beta emitters. After the measurement the samples are archived. In 2001 a number of filters, which were collected soon after the Chernobyl accident, were retrieved from the archive and analysed for ^{137}Cs with scintillation gamma spectrometry.

Methods and results

The filter material was either paper (Whatman 42) or glass microfibre (Whatman GF/A), with weekly air volumes about 1000 m^3 and 5000 m^3, respectively. Usually the filters are changed once a week, but due to a strike in Finland some filters were not changed on time. Due to the same reason, air flow readings were not available in some cases, and they were estimated based on

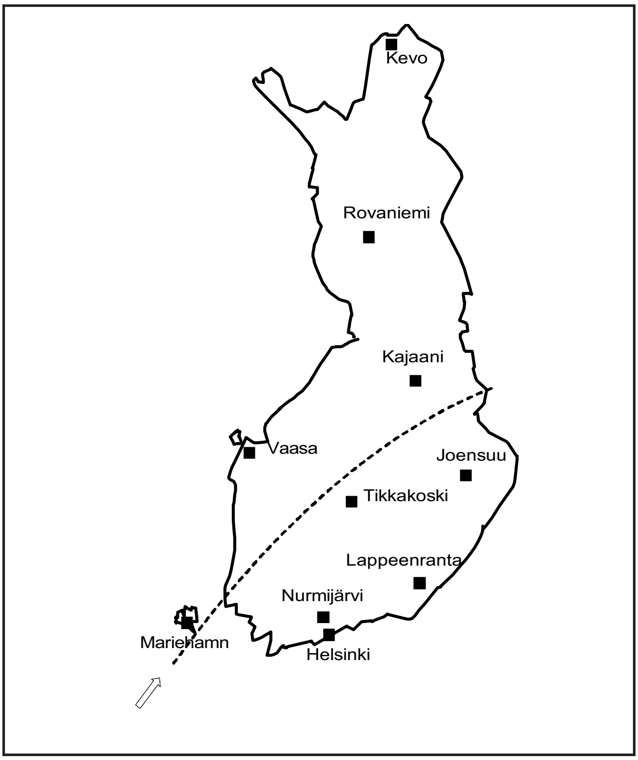

Figure 1. Location of sampling stations. The calculated path of the radioactive plume has been marked with an arrow and a broken line (Valkama *et al.*, 1995).

air flow of longer period than a week. In summer 2001 the filters collected during April–June 1986 were retrieved from the archive and analysed for ^{137}Cs with a scintillation gamma spectrometer with 4" x 2" NaI(Tl) detector. Measuring time was 24 hours, and background filter samples were measured during weekends. The detection limit was about 30 µBq/m^3.

The observed weekly ^{137}Cs activity concentrations in the air are presented in logaritmic scale in Figure 2. The one sigma counting error varied between 0.5 and 5.6 per cent. The values below detection limit were plotted as equal to the detection limit (= 30 µBq/m^3). The weekly concentration values decreased with an effective half-life of 4 – 9 days.

Conclusions

Trajectory calculations show that the initial Chernobyl plume arrived to Finland from south-west (Figure 1). Aircraft measurements indicate that the main part of the plume arrived at an altitude of 1000 – 2000 m (Sinkko *et al.*, 1987).

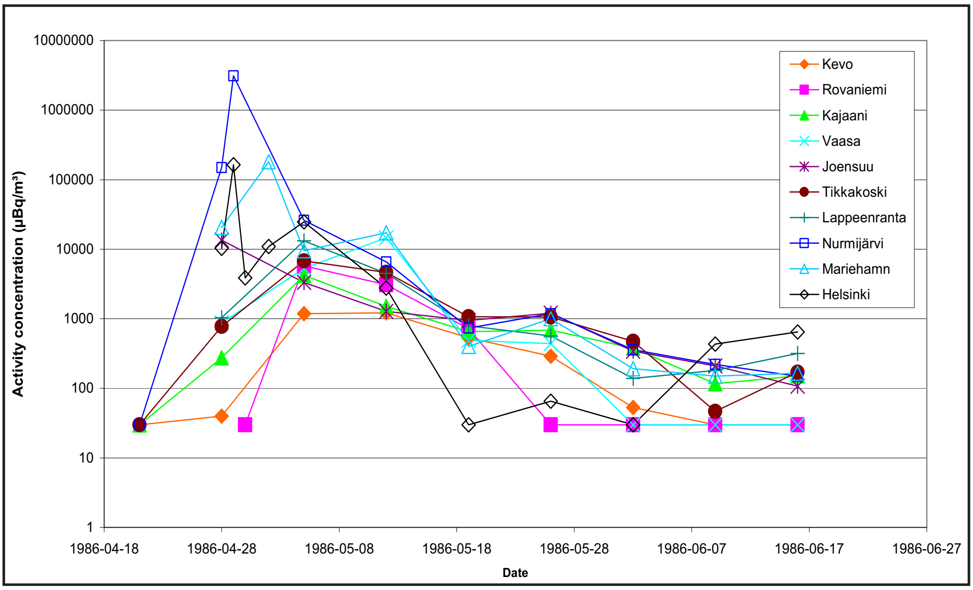

Figure 2. Cs-137 activity concentration in ground level air in Finland after Chernobyl accident.

A weather front located at the Kajaani latitude blocked the way of the plume to northern Finland which explains the relatively low concentrations found at Kajaani, Rovaniemi and Kevo (Figure 2). The highest ^{137}Cs activity concentrations in ground level air during the first few days after the accident were found at stations of Nurmijärvi, Mariehamn, Helsinki and Joensuu. The concentration was an order of magnitude lower at Helsinki compared to Nurmijärvi although the distance between these stations is only about 40 km. The cold surface of the Baltic sea cooled the surface air over the sea and coastal areas creating an inversion layer that the Chernobyl plume was unable to penetrate. The inversion layer did not reach Nurmijärvi 40 km inland and 105 m above sea level. Thus the afternoon convective flows brought the Chernobyl activity close to the ground. The inversion layer also protected ground level air at Mariehamn where at first only hot particles were observed. These particles were large enough to be able to cross the inversion layer due to the gravitational settling (Mattsson and Hatakka, 1987). The results of this study agree reasonably well with other studies. For example, Sinkko *et al.* (1987) has found that the maximum concentration was observed in the evening of April 28 at Nurmijärvi. Concentrations declined rapidly because of the change in weather conditions in Finland and because the direct transport of air masses from Chernobyl to Finland ceased on April 29 (Arvela *et al.*, 1987). The increase in ^{137}Cs concentration after 1986-06-07 might be related to resuspension after the melting of snow and beginning of agricultural activities and to air masses coming from heavily contaminated regions in eastern Europe.

The aerosol particles collected onto filters might be affected by the 15 years long storage. Thus the most volatile radionuclides might have partially evaporated. However, this seems not to be a problem in the case of ^{137}Cs which has been successfully measured even after almost 40 years of storage (Aaltonen *et al.*, 2002). Long-term collection and storage of environmental samples is still important as in most cases a retroactive sample collection is not possible.

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11d A survey of Cs-137 and Sr-90 in the lower part of the stream Verkmyån, Sweden

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Introduction

In the beginning of the 1990th airborne gamma spectrometry measurements identified local geographical anomalies in the ^{137}Cs load in the Gävle region in Sweden (Figure 1). The process behind and prevalence of these anomalies is not yet known. One of these areas, situated in Utnora 10 km north of the town Gävle, is the subject for a closer study (Tjärnhage *et al*. 2000, Nylén *et al.* 2001, Stark 2001). In this area the load of ^{137}Cs is approximately 10 times higher than in surrounding areas that received a deposition of >100 kBq/m^2 of ^{137}Cs.

The area is situated in an outflow area along the stream draining the lake "Hille" and is surrounded by coniferous forest. The upstream part of the

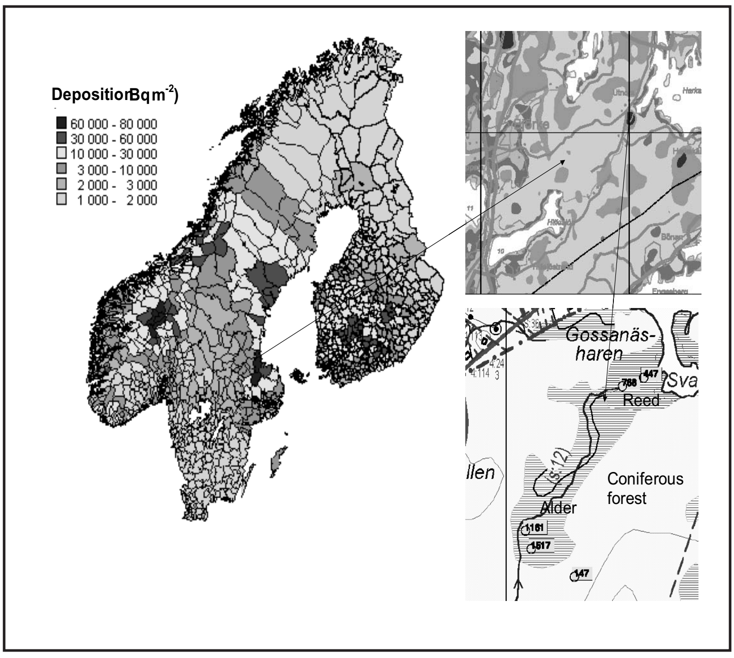

Figure 1. The study area.

outflow area consists of an alder forest that is generally flooded in the spring, while the down stream part, that ends in a nearby estuary in the Baltic Sea (2.5 km downstream lake Hille), consists of a belt of common reed (figure 1). Theses types of wetland constitute transition zones between terrestrial and aquatic systems and serve as nurseries and feeding areas for aquatic and terrestrial fauna. The high concentrations of ^{137}Cs may be a stress factor for the endangered groups of animals e.g. frogs, in this area.

A project with the aim to study the dynamics and exposition in the area is conducted as cooperation between the Swedish Radiation Protection Authority (SSI) and the Swedish Defence Research Agency (FOI). The part of the project presented here describes and compares the distribution of ^{137}Cs and ^{90}Sr in the hill slopes and out stream areas.

Experimental

Samples have been taken in two different out stream areas along Verkmyrån and from the adjacent hill slopes:

- An alder forest consisting of mature alder trees with an understorey of grass and herbaceous plants. The forest is flooded during the spring flood.
- A downstream reed area that is more or less constantly saturated by water and situated at the outlet of the stream
- The coniferous hill slopes higher up in the terrain surrounding the wetland and not in direct contact with the stream.

The area has been scanned with “Back Pack” field GPS-Gamma measurements during 1998 to 2001 resulting in a clear overlap of vegetation types and ^{137}Cs load (Tjärnhage *et al*. 2000, Nylén *et al.* 2002). From the results of these measurements the positions for soil samples have been chosen to calibrate the field measurements and to investigate the depth distribution of ^{137}Cs and ^{90}Sr. For this purpose three replicate soil (ca 40 cm depth) and vegetation samples have been taken in each area described above. All samples were measured at the NBC-analysis laboratory at FOI (Nygren *et al.* 2002, Nylén *et al.* 2001). At the time for this abstract just one of the samples from the coniferous forest had been analysed for ^{90}Sr. Therefore these results are preliminary and will be further analysed in the future.

Results

Load of ^{137}Cs and ^{90}Sr

The region (Gävle) received the highest deposition of ^{137}Cs in Sweden (Moberg 1991).

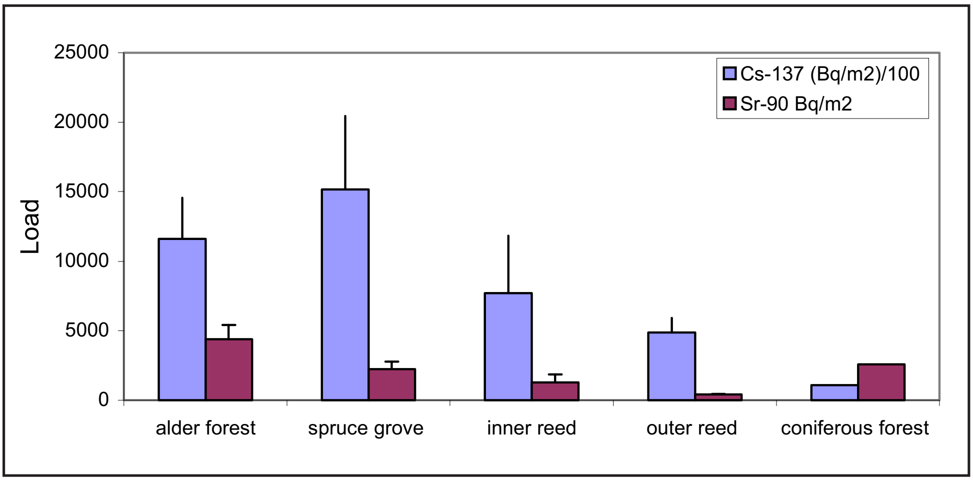

Figure 2. Load of ^{137}Cs and ^{90}Sr in the study area evaluated from the soil samples. The number of profiles analysed are three per area except for the coniferous forest where only one profile was evaluated. The lines above the bars are error bars.

The load of ^{137}Cs in the study area is elevated by a factor 7 to 15 in the alder forest and reed areas (table 1) compared to the hill slope that has values within the expected range in this region. The load of ^{90}Sr shows no such trend and is within the range expected from a Chernobyl fallout of ^{137}Cs of ca 100-200 kBq/m^2. A small spruce groove of 5x5 m in the upstream part of the alder area has the highest load of ^{137}Cs detected (1.5 MBq/m^2). A trend of decreasing load towards the sea is observed for both ^{137}Cs and ^{90}Sr. According to these measurements ^{137}Cs has been accumulated only in the areas that are flooded during the spring peak, while ^{90}Sr shows no such accumulation. Further more the accumulation seems to be more effective in the upstream part of the out stream area.

Table 1. The ratio between ^{90}Sr and ^{137}Cs. One sample profile per area is evaluated.

Area	**Ratio**[1] $^{90}Sr/^{137}Cs$	**S.D. %**
Alder forest	0.003	25
Inner reed	0.002	48
Outer reed	0.002	6
Coniferous forest	0.02	2

$^{90}Sr/^{137}Cs$ ratios

The calculated ratios between ^{90}Sr and ^{137}Cs in one soil profile per area are presented in table 1. Only the coniferous forest, that has the ratio 0.02, falls within the range 0.01-0.03 expected from the Chernobyl fallout (*Moberg et al, 1991)*, while the other areas as a result of the high load of ^{137}Cs show lower ratios.

Vertical distributions of ^{137}Cs and ^{90}Sr

Although the geographical distribution differs between the two nuclides and ^{137}Cs seems to have been more effectively accumulated the soil profiles show resemblance between the vertical distribution of ^{90}Sr and ^{137}Cs (figure 3). There is no clear trend of higher mobility for ^{90}Sr and the peak is in both cases about 5 cm below the soil surface.

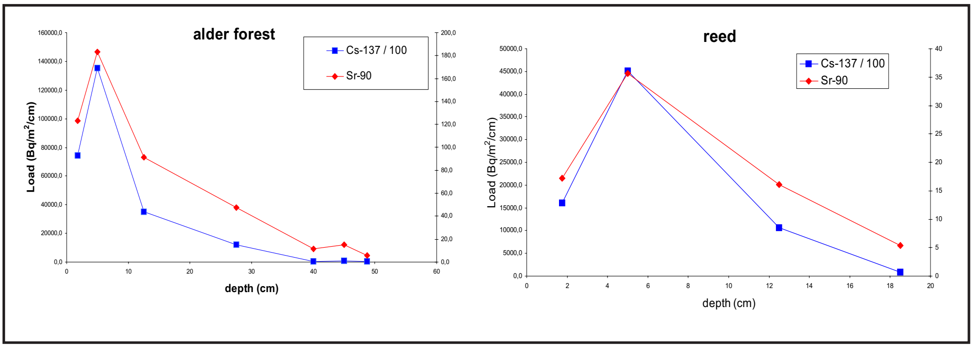

Figure 3. Examples of vertical distributions in the alder forest and the reed area.

Discussion

Flooding during the spring 1986 followed by an effective retention of ^{137}Cs has most likely caused the high load of ^{137}Cs in the alder forest and the reed areas. A sharp gradient in load of ^{137}Cs between the alder forest (high load) and the adjacent and topographically elevated coniferous forest (low load) determined by a backpack technique also supports this hypothesis (Tjärnhage etal, 2000). The deposition in the coniferous forest is consistent with what is measured in other places around the coastal area. Hence it is likely that the accumulation in the alder forest and reed area must be due to redistribution from the lake "Hillesjön" via the stream Verkmyrån. The main contribution to the elevated values in the alder forest and reed areas should have occurred during the spring flood in 1986 when a main fraction of the intercepted fallout still was in the water phase of the lake (Moberg, 1991). It is likely that the contribution during later years have been less significant. Larger fallout from air in this narrow area in the outlet from Verkmyrån is less likely to be the explanation.

The Load of ^{90}Sr in the outflow areas seems to be far lower than of ^{137}Cs. This could be due to a less effective interception in the out stream areas during flooding or more effective interception of ^{90}Sr in the lake Hillesjön preventing a drainage of ^{90}Sr via the stream. A slower process of less effective retention in soil over the years elapsed after the Chernobyl accident is not supported by the similar depth distribution between the two nuclides.

This study will continue with a more detailed mapping with nuclide specific and TLD field measurements, calculations of exposition in different sub areas and budget calculations for ^{137}Cs.

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11e Radionuclides in algaes from the bay just north of Barsebäck nuclear power plant

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Abstract
The activity concentration of Cs-137, I-131, Zn-65, Co-60, Co-58, Mn-54, and K-40 were determined in samples of brown seaweed *(Fucus)* using high-resolution gamma-spectrometry. The algaes were sampled in the bay just north of the Barsebäck nuclear power plant (55.4 N, 12.6 E) in the south of Sweden to study the variation in contamination levels in the shallow waters. Significant levels of the typical neutron activation products Zn-65 (up to 3 Bq/kg dw), Co-60 (100-300 Bq/kg dw), Co-58 (1-7 Bq/kg dw) and Mn-54 (12-27 Bq/kg dw) were found in the algae samples. The results were compared with those of similar studies at the same location performed 25 years ago. The current concentrations were found to be considerably lower especially for Zn-65 and Co-58. The activity concentration of gamma emitting radionuclides in *Fucus* from the bay just north of Barsebäck is today dominated by (in order of decreasing concentration): natural K-40, Co-60 from the plant, Cs-137 from Chernobyl, Mn-54 and Co-58 from the plant. Further studies will show if some parts of the bay are suitable for studies of absorbed doses and possible effects on various biota.

Keywords: Barsebäck NPP, Fucus, activation products, marine environment.

Introduction
The Barsebäck nuclear power plant (Barsebäck NPP) is one of four nuclear power plants that provide Sweden with 44% of its annual electricity consumption. Until recently, the plant consisted of two reactors in operation, Barsebäck-1 and Barsebäck-2, both of boiling water type, having an electrical effect of 600 MW, or 1700 MW thermal powers each. The location of the plant is controversial since it is located 30 km away form the Danish capital, Copenhagen and has in total around 2 million inhabitants within a distance of 30 km. At the end of November 1999, Barsebäck-1 was shut down, while Barsebäck-2 is still in operation.

During normal operation of a nuclear power plant, small controlled releases mainly of neutron activation products are common and will occur through the cooling water outlet. There are also external sources of Cs-137 and I-131 in the marine environment in the form of debris from the Chernobyl accident (Cs-137) and releases from hospitals and patients (I-131). In the case of Barsebäck, currents through the Öresund transport Cs-137 from the Chernobyl fallout in the Baltic Sea to the area. Furthermore, Cs-137 from reprocessing plants (e.g. Sellafield) and from the global fallout from nuclear weapons tests in the atmosphere have also been shown to contribute to the Cs-137 concentration in Swedish coastal waters [1].

Levels of various radionuclides in the vicinity of Barsebäck NPP are followed by a monitoring programme conducted by the Swedish Radiation Protection Authority (SSI) [2]. This study will in more details investigate how the radionuclides are distributed in the near vicinity of the plant. Even though the discharged radionuclides from the Barsebäck NPP are dispersed into large volumes of water, the bio-indicator used (*Fucus*) has the ability to accumulate and concentrate several of the radionuclides of interest [3,4,5] into its biomass.

The results are compared with those of a similar research performed in 1977, but with longer distances between sampling sites at that time. It is also aimed at investigating the relation between the activity concentration in *Fucus* and in other organisms in the bay.

Material and Methods

The location where collection has been made is situated around 1 km north of the discharge pipe (Fig.1.)

Samples of *Fucus* were collected in May 13, 2002 from water depths around 0.5 m. During sampling, the wind speed was slow and the waves were mild. The samples were collected from different distances from the Barsebäck NPP (see fig.1.). These samples were dried in room temperature for 5 days. Then the samples were ground, packed in 150 ml plastic beakers, and weighed.

Table.1. Sample ID and the sampling distances from the Barsebäck NPP cooling water discharge pipe.

Sample ID	B5	B6	B7	B8	B9	B10	B11	B12	B13	B14
Distance from the point of release (m)	780	880	960	1060	1130	1190	1210	1280	1310	1440

The activities of Cs-137, I-131, Zn-65, Co-60, Co-58, Mn-54 and K-40 were determined by gamma-ray spectrometry with acquisition times between 100,000-150,000 s. The measurements were performed using HPGe (EG&G Ortec) detector of 36% efficiency, and energy resolution of 1.8 keV at 1.33 MeV. The spectrometer was calibrated by sources of known activity. The accuracy of the measurements has been evaluated through an inter-comparison with sources prepared by Risoe National Laboratory, Denmark.

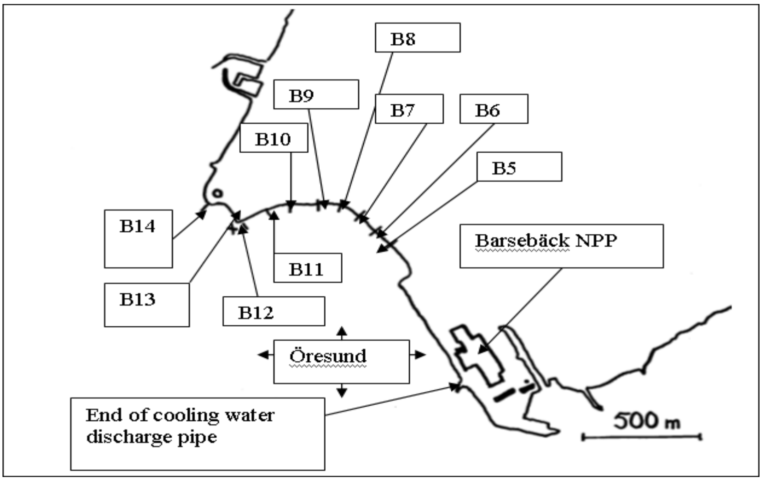

Fig.1. Map of the Barsebäck NPP and the spots where samples of algae were taken.

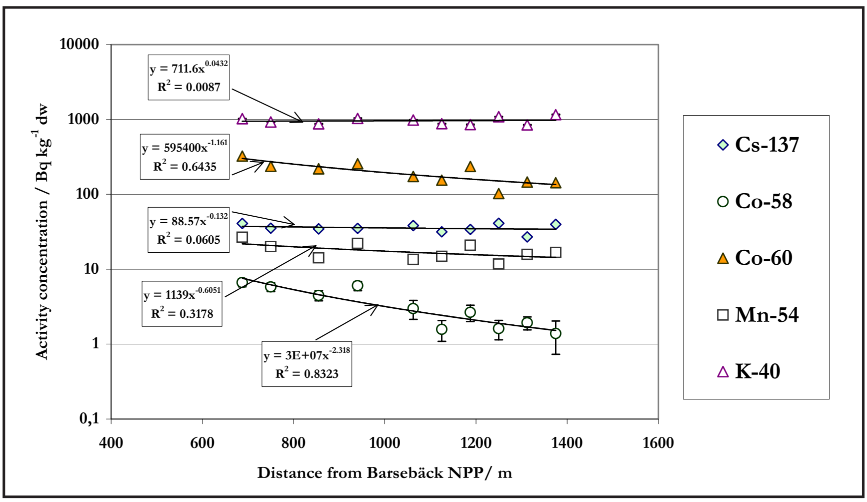

Fig.2. Activity concentration in Fucus in relation to the distance from the outlet of Barsebäck NPP in May 2002. Power functions were fitted to the observed data. Error bars indicate the uncertainty of the pulse statistics (1 Standard deviation).

Results and discussion

The activity concentrations of Cs-137, Co-60, Co-58, Mn-54, K-40 in the *Fucus* samples in relation to the distance from the outlet of Barsebäck is given in Fig.2.

The results (Fig.2.) show that the activity concentrations of Co-60, Co-58, and Mn-54 decrease with distance from the point of release and that the dependence of distance can be described by a power function as in 1977 [5]. The main part of the Cs-137 found is most likely from Chernobyl fallout over the Baltic. In some of the samples, significant concentrations of I-131 were found.

Table 2. Comparison of variation of main concentration levels of radionuclides in June 1977 and May 2002.

Radionuclide	Physical half-life	Radionuclide concentration, June 12, 1977 (Bq/kg dw)	Radionuclide concentration, May 13, 2002 (Bq/kg dw)
Zn-65	244 days	150	1.8
Co-60	5.27 years	1000	199
Co-58	71 days	500	3.5
Mn-54	312 days	100	17.6

Comparison with previous results

The average activity concentrations of neutron activation products in the *Fucus* samples in the bay are compared with the results that were obtained in the year of 1977 (Table 2). It is evident that all activity concentration levels in 2002 are significantly lower than those in 1977.

The data are also compared with the values obtained by the SSI in the year 2000 [2]. For Cs-137, which does not exhibit any distance dependence, the data is in fair agreement.

Conclusions

The activity concentration of gamma emitting radionuclides in *Fucus* from the bay just north of Barsebäck is today dominated by (in order of decreasing concentration): natural K-40, Co-60 from the plant, Cs-137 from Chernobyl, Mn-54 and Co-58 from the plant. The concentration of Barsebäck-produced radionuclides is lower per MW_e than in 1977.

Acknowledgement

This project was partially funded by The Swedish Institute (SI).

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11f Long-term behaviour of ^{90}Sr and ^{137}Cs in surface water in Finland

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Introduction

Activity concentrations of two long-lived radionuclides, ^{90}Sr and ^{137}Cs, have been determined from water of five large rivers discharging from the mainland to the Gulf of Finland and Gulf of Bothnia since 1965, and of several lakes and rivers in various large drainage areas in Finland since 1986. In the 1960s those radionuclides originated from athmospheric nuclear weapon tests. In the 1970s and 80s small peaks in surface water were caused by the Chinese nuclear weapon tests. The depositions of ^{137}Cs and ^{90}Sr during the nuclear weapon test period were distributed almost evenly in Finland. $^{137}Cs/^{90}Sr$ in deposition varied within 1.4 ± 0.2 at various survey stations. The Chernobyl depositions of ^{137}Cs and ^{90}Sr, on the contrary, were most unevenly distributed in Finland.

Material and methods

Water samples from the mouths of five rivers, Kymijoki, Kokemäenjoki, Oulujoki, Kemijoki and Tornionjoki since 1965, and from several lakes and rivers in each of the large drainage areas 1-11 (Fig. 1) in 1986-1992, were analysed for ^{137}Cs and ^{90}Sr (1-2).

Results and discussion

In the fallout period the ^{90}Sr content in water was clearly higher than that of ^{137}Cs. ^{137}Cs was lowest in the Kokemäenjoki river. Since the deposition was almost the same in all the areas, the differences in the activity concentrations in water from the river mouths reflect the impact of the catchment area on the behaviour of these nuclides.. The abstention of ^{137}Cs in the catchment area of the Kokemäenjoki river was evident. The increase of ^{137}Cs and ^{90}Sr due to the Chernobyl deposition was highest in the Kokemäenjoki and Kymijoki rivers and much slighter in the other rivers. The increase of ^{90}Sr was negligible in Kemijoki and Tornionjoki (Fig. 2).

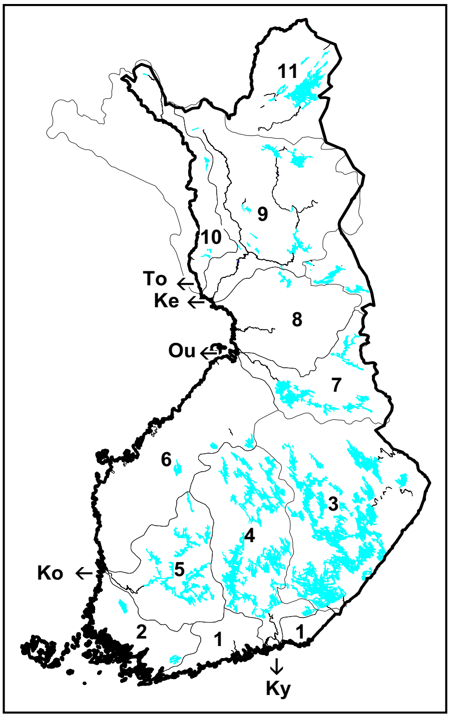

Fig. 1. Drainage basins of Finland (areas 1-11) and the rivers Ky= Kymijoki, outlet river of area 4; Ko=Kokemäenjoki, outlet river of area 5; Ou=Oulujoki, outlet river of area 7; Ke=Kemijoki, outlet river of area 9; To=Tornionjoki, outlet river of area 10.

After the Chernobyl deposition ^{137}Cs in water decreased more rapidly than ^{90}Sr. During 1986-1988 activity concentrations of ^{137}Cs decreased from the top values by 84% in the Kymijoki and Kokemäenjoki rivers, and ^{90}Sr by 33% and 41%, respectively. Already in 1994, and thereafter, activity concentrations of ^{90}Sr have been higher than those of ^{137}Cs in the rivers of northern Finland.

Ecological half lives of ^{137}Cs and ^{90}Sr detected in the rivers during the fallout period 1965-1985 were almost the same for these nuclides, that is 7-8 years. The ecological half life for ^{90}Sr was longest (about 12 y, in 1989-2001) in the Kemijoki and Tornionjoki rivers, where the increase by Chernobyl deposition was low. In 1988-2001 ^{137}Cs decreased more rapidly than ^{90}Sr in all the rivers, their ecological half- lives being 4.2-6.5 y. Because activity concentrations of ^{90}Sr in the Kemijoki and Tornionjoki rivers stayed almost unchanged due to the Chernobyl deposition, ecological half times for ^{90}Sr were also estimated for the whole time period 1965-2001. They were approeximately 12 y and 10 y for Kemijoki and Tornionjoki, respectively. (Fig. 2).

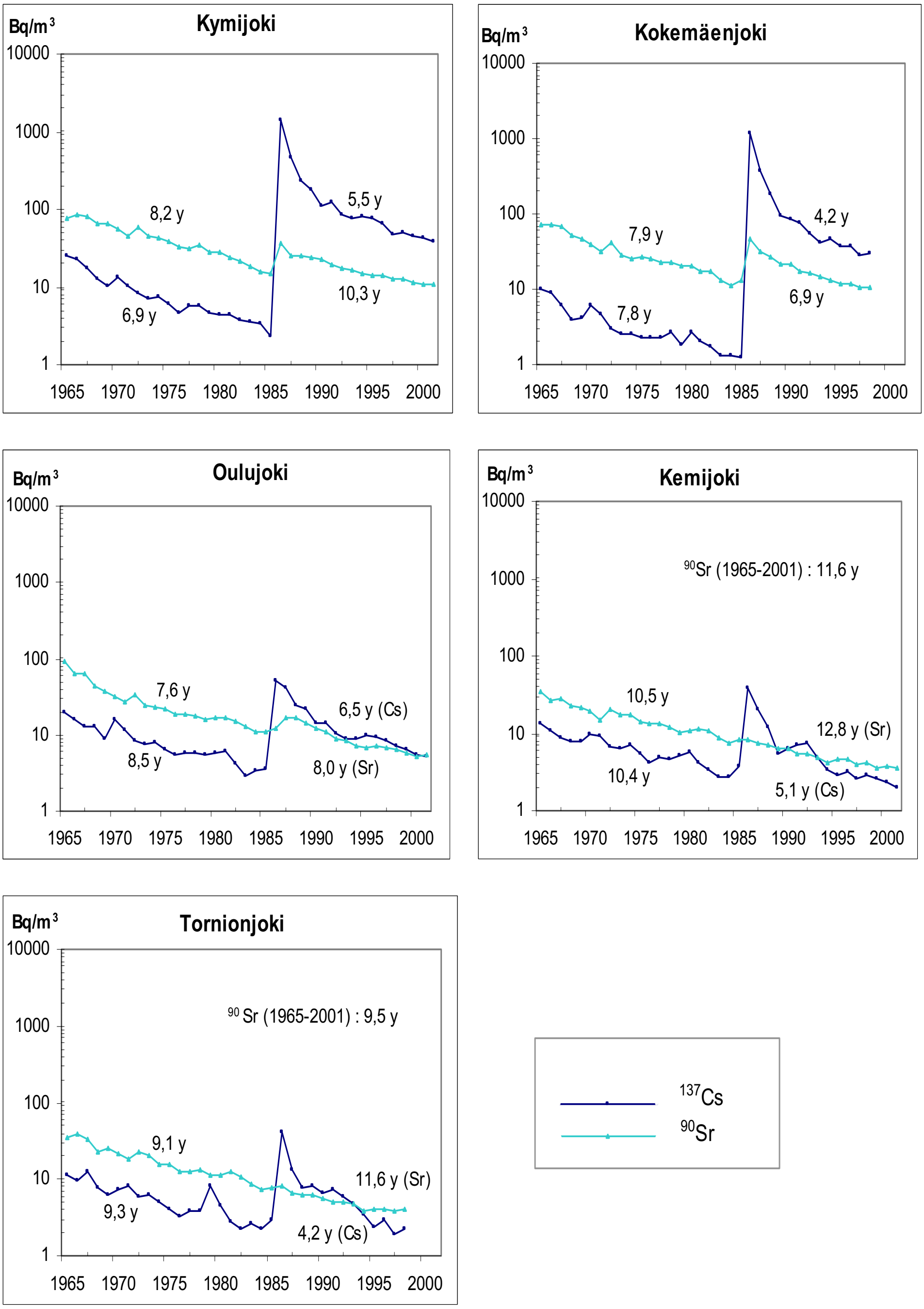

Fig. 2. Activity concentrations of ^{137}Cs and ^{90}Sr (Bq/m^3) in Kymijoki, Kokemäenjoki, Oulujoki, Kemijoki and Tornionjoki rivers in 1965-1998 or 1965-2001. Ecological half times (y) for ^{137}Cs and ^{90}Sr in 1965-1985 and 1989-2001are also given in the figure.

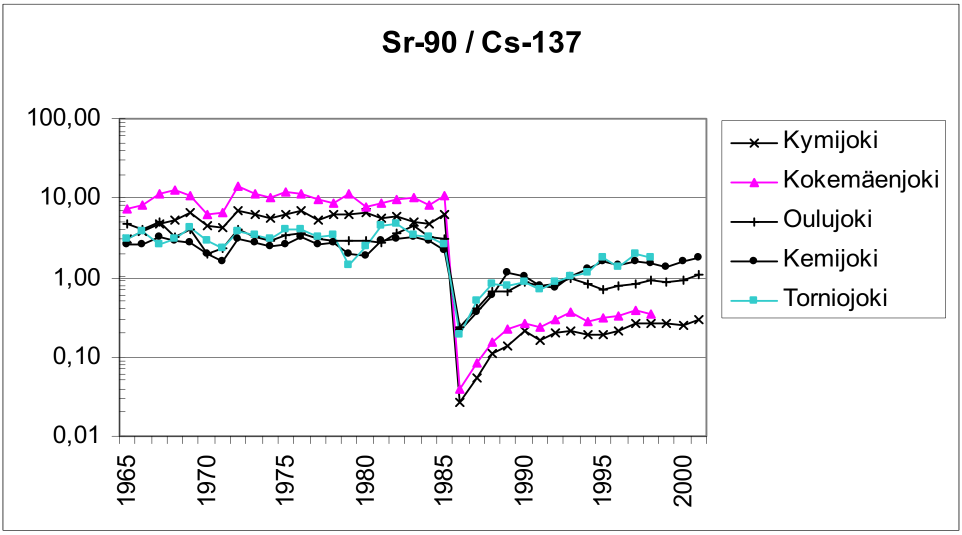

Fig. 3. $^{90}Sr/^{137}Cs$ in the water of Kymijoki, Kokemäenjoki, Oulujoki, Kemijoki and Tornionjoki rivers in 1965-1998 and 1965-2001.

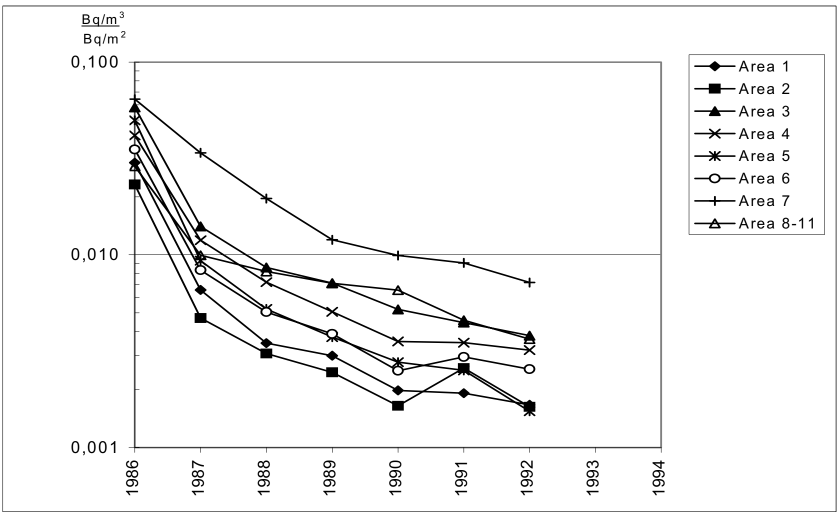

Fig. 4. Average transfer coefficients for ^{137}Cs from deposition to surface water (TF_w = (Bq/m^3 in water)/(Bq/m^2 deposited)) in various drainage areas. The areas are given in Fig. 1.

The ratio ^{90}Sr / ^{137}Cs was highest (about 10) in the fallout period in Kokemäenjoki, then decreased due to Chernobyl accident to 0.04 and is now increasing gradually. In the Kemijoki and Tornionjoki rivers the ratio is approaching the pre-Chernobyl value (Fig. 3).

After the Chernobyl deposition average transfer coefficients from deposition to lake and river water in the drainage areas differed from each other, being highest in the area 7 (Fig. 4).

Radiation dose from ^{137}Cs and ^{90}Sr in drinking water to people in southern Finland, where lake water from the area of highest Chernobyl deposition in Finland is used as raw water, was estimated to be 0.4 - 0.5 μSv in 2001. About 44% of this dose was caused by ^{90}Sr and 56% by ^{137}Cs. The relative contribution of ^{90}Sr increases the longer the time after the deposition is, because ^{90}Sr is, differently from ^{137}Cs, only slightly removed from water by natural processes and not at all in the water treatment of waterworks.

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11g Radiocaesium in NE Estonian Soil

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Our first countrywide study carried out in 1991-1993 determined both geographical and depth distributions of the deposited ^{134}Cs and ^{137}Cs in the Estonian soil. According to the results of this study and an independent air-borne scanning, radiocaesium originating from the 1986 Chernobyl accident with the mean deposition of 2 kBq m^{-2} was extremely unevenly distributed over the country [1,2]. The mapped distribution is presented in Fig. 1. Our estimates demonstrated that about 2/3 of the total ^{137}Cs inventory in Estonia was deposited in Ida-Virumaa County, NE Estonia, where the maximum depositions reached 40 kBq m^{-2}.

The re-gion was revisited for soil sampling in 1998-2001. Undisturbed soil profiles down to a depth of ~ 20 cm were collected and the 2-3 cm sample core slices analyzed using a low-back-ground HPGe gamma spectrometer (42% efficiency and 1.7 keV resolution). As in the previous study, both total depositions and depth distributions of the deposited ^{134}Cs (where possible) and ^{137}Cs activity concentrations were determined. Depth distributions of the deposited activity showed considerable site-specific variations. In addition, it appeared that in comparison with our previous study, clear features of time-dependent migration to deeper soil layers were evident.

The preliminary modeling has demonstrated that a lognormal distribution with varying parameters fits satisfactorily the determined depth-distributions of the Chernobyl radiocaesium activity concentration. An attempt was made to apply a single model to describe the observed time-de-pendent depth-distribution pattern over the time period of 1986-2001. We started from the multi-compartmental migration model for undisturbed soil [3] (here Model RP72), presented in Fig. 2. This model has been elaborated bas-ing on experimental migration results for Pu for a time period up to 30 y. For our application, the model is modified to account for the radioactive decay of ^{137}Cs. The results of modeling using the original transfer parameters are given in Fig. 3 (a).

A numerical analysis of the model confirms that, while the lognormal depth-distri-bution of activity concentration is correctly described, the averaged transfer coefficient values, derived for Pu migration in [3], should be slightly modified to fit our results: $w\sim 0.03t$ and $xc\sim 0.3t$ (Fig. 4). The calculated

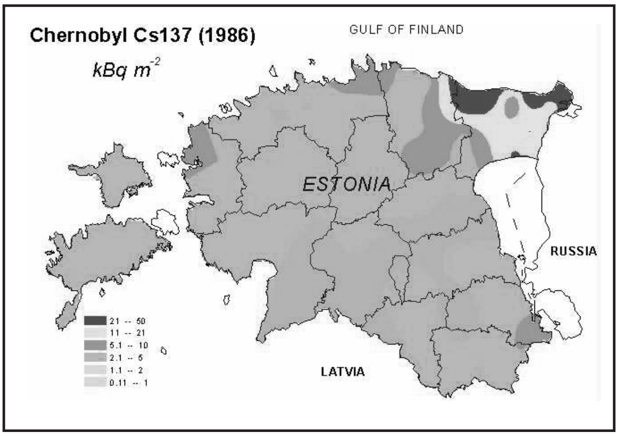

Fig. 1. Cs137 from Chernobyl fallout (1986).

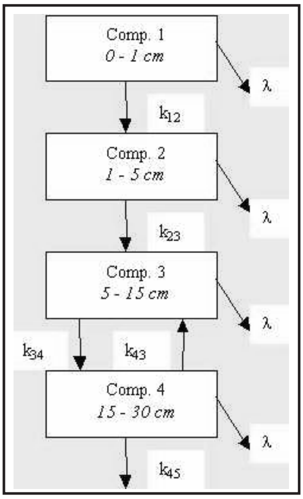

Fig. 2. Compartmentl model RP72 [3].

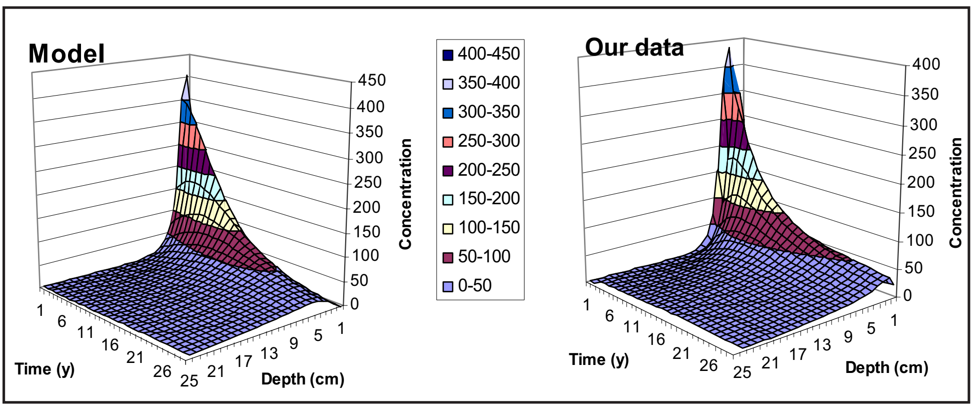

Figure 3. Time-dependent depth distributions for 137Cs: (a) Model PR72 [3] and (b) the model fitting our data.

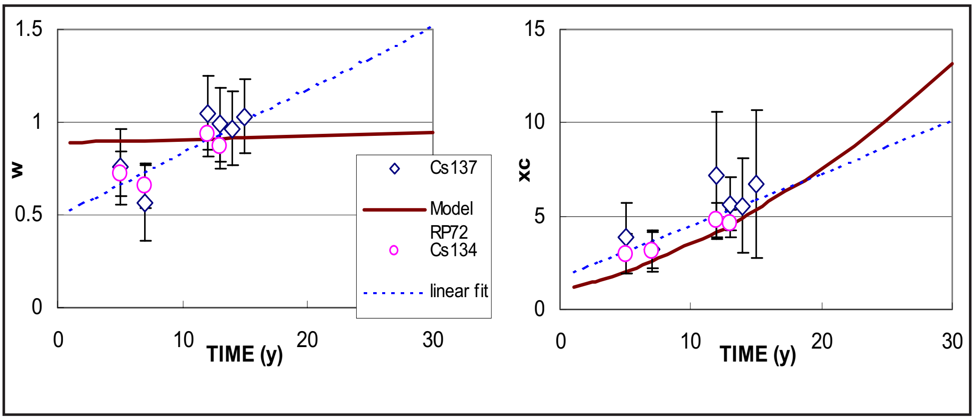

Figure 4. Time-dependent parameters *w* and *xc* of lognormal depth distribution for two models.

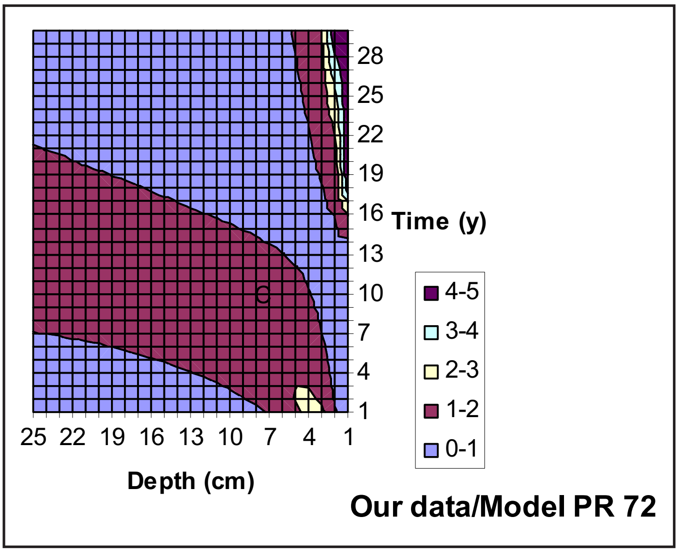

Figure 5. A ratio of 137Cs activity concentrations calculated using two models. models.

model fitting our measured results is presented in Fig. 3 (b). The ratio of time- and depth-dependent activity concentrations obtained using these two models is given in Figure 5.

In general, a slow migration rate and consequently a strong effective retention of radiocaesium in upper soil layers follow from the found relatively weak time-dependence of the fitting parameters.

The authors acknowledge a partial support by the ESF grants.

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11h Migration of ^{137}Cs in Swedish soil profiles

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Introduction

The study comprises eight grass field sites in that part of Sweden, which received high ^{137}Cs fallout from the Chernobyl accident in 1986. The soils, six mineral soils and two peat soils, were sampled to certain depths during the years 1987-2000, one of the peat soils also in 2002 (Rosén et al., 1999; Hermansson, 2000). The aim was to study the migration of ^{137}Cs in the soil profiles since 1986 and relate this migration to soil characteristics. During the first years the effect of K-status on plant uptake of ^{137}Cs was also studied.

Materials and methods

The sites received 16- 200 kBq per m^2 of ^{137}Cs in 1986 (Rosén et al., 1999) and have not been ploughed after the fallout event. Some data to describe the sites are given Table 1.

Sampling procedure

On each site 3 circle formed plots with 5 m radius were selected (A, B and C), as shown in the figure below. In each plot 5 soil cylinders were taken, the surface 10-cm soil with a conventional auger, diameter 5.7 cm, the next 10-60 cm with a special Ultuna auger, diameter 2.2 cm. During 1987- 1995 samples were also taken down to 25 cm, and during 2000-2002 down to 60 cm on all the sites. The soil cores were divided in 1-cm layers down to 10 cm and then in 2.5-cm layers down to 60 cm.

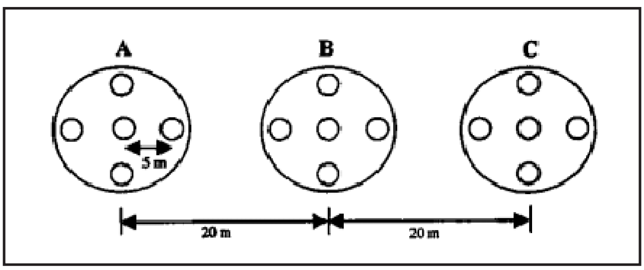

Table 1. Description of sites, sampling times and deposition of ^{137}Cs in 1986

Sampling site	Longitude and Latitude	Altitude, m	Sampling year	Precipitation, yearly, mm 1961-1999	Soil type	Soil taxonomy, FAO-legend 1988	Deposition 1986, mean, kBq/m^2
Hille	60°45´N 17°13´E	20	1987 1989 1994 2000 2002	617	Peat	Histosol	200,1
Möjsjövik	59°13′N 17°15′E	20	1987 1994 2000	650	Peat	Terric Histosol	70,5
Skogsvallen	60°10′N 17°11′E	40	1987 1992 1994 2000	566	Silty clay	Dystric Cambisol	92,4
Ramvik	62°50′N 17°49′E	25	1987 1994 2000	703	Silt loam	Eutric Regosol	62,3
Hammarstrand	63°06′N 16°22′E	125	1989 1994 2000	539	Silt loam	Eutric Regosol	45,3
Blomhöjden	64°36′N 14°44′E	460	1989 1994 2000	754	Sandy loam	Typic Haplocryod*	15,7
Stora Blåsjön	64°51´N 14°08´E	480	1989 1994 2000	699	Sandy loam	Dystric Regosol	37,8
Åsenmossen	60°35´N 17°14´E	75	2000	563	Peat	Fibric Histosol	43,5

*Soil taxonomy acc.Soil Survey Staff 1992,

Prior to soil sampling the grass vegetation was cut in each plot, 0.25-1.0 m^2 in size. Totally 3-5 grass samples were cut 5 cm above the soil surface, dried at 90 °C, ground and pooled to-gether (A+B+C). The soil samples were dried at 40°C for at least 5 days, crushed in a mortar, passed through a 2 mm sieve and weighed. The dried plant and soil samples were transferred to plastic containers for activity analysis.

Calculations

Deposition and distribution in the soil profiles of ^{137}Cs-activity, Bq/m^2, are given in detail in appendix, from which Table 2 and Table 3 have been calculated to establish three deposition-independent parameters for comparison of migration between the sites.

Firstly the nuclide distribution was calculated as fraction or percent of the total ^{137}Cs-content found in each soil layer. Secondly the mean migration depths were calculated in two ways, as statistical median depth for the years 1987-1994 (Rosén, 1999) and as migration centre for the years 2000-2002 (Hermansson, 2001). The migration centre, (x, cm), defined as the gravity point of activity in the soil profile (Arapis et al., 1997), was calculated as

$$\sum_{i=1}^{n} (X - X_i) q_i = 0$$

where X_i (cm) is the depth from the surface to the centre of soil layer i and q_i the nuclide frac-tion in the same soil layer. Thirdly the migration rate (cm/year) was calculated by dividing, the median depth or the migration centre, by the number of years elapsed since the deposition in 1986. The latter, weighted method, gives slightly smaller values than the former, unweighted method, of mean depths and migration rates (Forsberg, 2000).

Results

Distribution of ^{137}Cs

Table 2 shows the distribution of ^{137}Cs-activity in the soil profile at different years. Generally the ^{137}Cs-activity was redistributed between 0-5 cm layer and the 5-25 cm layer with time up to 1994. Later a small part of the ^{137}Cs-activity migrated down below 25 cm. Largest redistribution was found at the Hille site, where the activity in the 0-5 cm layer changed from 90 % in 1987 to 40 % in 2002 and in the 5-25 cm layer from 10 % in 1987 to 59 % in 2002. On all other sites the ^{137}Cs-activity was highest in the 0-5 cm layer at the end of the observation period.

Table 2. Percent of total ^{137}Cs- activity, present in the soil layers 0-5, 5-25 and the layer 25-60 cm, at different years after fallout in 1986.

Depth, cm	Hille					Möjsjövik			Skogsvallen			
	1987	1989	1994	2000	2002	1987	1994	2000	1987	1992	1994	2000
0-5	90,0	87,6	49,5	42,0	40,3	91,8	81,3	68,7	97,7	96,3	91,5	82,9
´5-25	10,0	12,4	50,5	54,0	59,0	8,3	18,7	24,0	2,3	3,7	8,5	15,8
25-60	-	-	-	0,0	0,7	-	-	7,3	-	-	-	0,3

Depth, cm	Ramvik			Hammarstrand			Stora Blåsjön			Blomhöjden			Åsen-moss en
	1987	1994	2000	1989	1994	2000	1989	1994	2000	1989	1994	2000	2000
0-5	95,7	70,6	52,0	88,5	74,4	70,5	90,5	82,5	83,7	90,6	72,2	64,8	33,7
´5-25	4,3	29,4	47,3	11,5	25,6	25,3	9,5	17,5	12,1	9,4[a]	27,8	25,7	53,6
25-60	-	-	0,7	-	-	4,2	-	-	4,2	-	-	9,5[b]	12,7[b]

a) 5-10 cm, b) 25-40 cm

Mean depth and migration rate

Table 3 shows the mean depth (cm) and migration rate (cm/year). Largest mean depth was found at Åsenmossen, 9,3 cm, and smallest at Skogsvallen, 3,1 cm. On all the sites the initial migration rate was highest at first registration, 1987 or 1989. Highest migration rate in 1987 was found at Hille and Möjsjövik, 1.0 cm/year and lowest at Skogsvallen, 0,6 cm/year.

Table 3. Mean depth, (cm), and migration rate (cm/year), for the years 1987-2002.

Sampling site	Soil type	Mean depth, cm					
		1987	1989	1992	1994	2000	2002
Hille	Peat	1,0	1,5	-	5,1	7,1	7,5
Möjsjövik	Peat	1,0	-	-	2,6	5,8	-
Skogsvallen	Silty clay	0,6	-	0,9	2,3	3,1	-
Ramvik	Silt loam	0,9	-	-	3,5	7,1	-
Hammarstrand	Silt loam	-	1,7	-	3,3	5,5	-
Stora Blåsjön	Sandy loam	-	1,4	-	2,7	3,4	-
Bjomhöjden	Sandy loam	-	-	-	2,4	6,0	-
Åsenmossen	Peat	-	-	-	-	9,3	-
Sampling site	**Soil type**	**Migration rate, cm/year**					
		1987	1989	1992	1994	2000	2002
Hille	Peat	1,0	0,2	-	0,6	0,5	0,4
Möjsjövik	Peat	1,0	-	-	0,2	0,8	-
Skogsvallen	Silty clay	0,6	-	0,1	0,2	0,3	-
Ramvik	Silt loam	0,9	-	-	0,4	0,6	-
Hammarstrand	Silt loam	-	0,6	-	0,3	0,4	-
Stora Blåsjön	Sandy loam	-	0,5	-	0,2	0,1	-
Bjomhöjden	Sandy loam	-	-	-	0,3	0,7	-
Åsenmossen	Peat	-	-	-	-	0,8	-

Discussion

During the years 1986 to 2002 the ^{137}Cs-activity migrated downward in all the eight soil pro-files examined. During the years 1986-1994, with exception for the site Hille, the lowest ^{137}Cs-activity was found in the layers below 5 cm. This shows that ^{137}Cs migrate slowly with time. Table 2 shows that in 1987 as much as 90 % or more of the total activity was found in the 0-5 cm layer. After 14-16 year ^{137}Cs had migrated downwards, more on the three organic soils than on the five mineral soils.

Mean depth and migration rate of ^{137}Cs varied between the sites. In the year 2000 the mean depth was below 5 cm on all sites except Skogsvallen and Stora Blåsjön, Table 3. With respect to migration rate the sites can be divided in two groups. The lowest group, 0.1-0.4 cm/year comprised the sites Skogsvallen, Hammarstrand and Stora Blåsjön, all three mineral soils. The highest group, 0.6-0-8 cm/year, comprised the organic soils Hille, Möjsjövik, and Åsenmossen and the sandy soils Ramvik and Blomhöjden.

Some of the ^{137}Cs deposited in 1986 after the Chernobyl was taken up by the grass vegetation. As shown earlier (Rosén, 1996), the uptake was highest the first years after fallout.

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Appendix Table. Distribution of ^{137}Cs in soils, kBq/m^2, for different years. The 25-60 cm layer was only registrated in 2000 and 2002. Values for the years 1987-1995 are from Rosén et al, (1999).

Depth, cm	*Hille*					*Möjsjövik*			*Skogsvallen*			
Year	**1987**	**1989**	**1994**	**2000**	**2002**	**1987**	**1994**	**2000**	**1987**	**1992**	**1994**	**2000**
0-5 cm	153,76	150,32	100,12	101,41	84,76	70,86	52,35	46,31	89,15	89,18	82,93	72,7
5-25 cm	18,08	18,39	99,27	131,46	93,12	6,38	11,91	16,14	2,15	3,40	7,39	13,9
25-60 cm	-	-	-	8,48	3,17	-	-	4,90	-	-	-	1,08
>2 mm	-	-	8,80			-	-	2,61	-	2,20	2,25	3,08
Total	171,84	168,71	208,19	241,35	210,48	77,24	64,30	69,96	91,30	94,78	92,57	90,8
Mean					**200,1**			**70,5**				**92,4**

Depth, cm	*Ramvik*			*Hammarstrand*			*Stora Blåsjön*			*Blomhöjden*			*Åsen-mossen*
Year	**1987**	**1994**	**2000**	**1989**	**1994**	**2000**	**1989**	**1995**	**2000**	**1989**	**1994**	**2000**	**2000**
0-5 cm	45,28	40,28	34,97	39,27	19,76	44,23	39,91	17,93	37,32	11,86	9,14	12,82	14,64
5-25 cm	2,72	16,48	45,87	5,43	6,62	15,88	5,15	3,53	5,39	1,24	3,51	5,02	23,27
25-60 cm	-		0,97	-	-	2,61	-	-	1,86	-	-	1,15a	5,54^{a}
>2 mm	-		0,34	0,86	1,02	0,12	-	-	2,27	-	1,61	0,73	-
Total	48,00	56,76	82,15	45,56	27,40	62,84b	45,06	21,46	46,84	13,10	14,26	19,72	43,45a
Mean			**62,3**			**45,3**			**37,8**			**15,7**	**43,5**

a) 25-40 cm

11i ^{137}Cs and ^{90}Sr in dairy and farm milk in Finnish Lapland 1960-2000

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Introduction

The decrease rates of ^{90}Sr and ^{137}Cs in milk were studied in the years following the atmospheric nuclear weapon testings and after the Chernobyl accident in 1986. The highest ^{90}Sr and ^{137}Cs concentrations in the 1960's in Finnish milk were recorded in Lapland even though the deposition of ^{90}Sr and ^{137}Cs did not significantly differ between other localities in Finland, and they were not greater in northern Finland. This was mainly due to the high proportion of peat soils and nutrient deficiency of the pastures in Lapland. ^{137}Cs deposition after the Chernobyl accident in 1986 in Lapland was less than 1 kBq m^{-2}, and ^{90}Sr deposition was so low that there was no detectable increase in the ^{90}Sr concentration in milk. The ecological half-lives were estimated for the decrease in the concentrations of ^{137}Cs and ^{90}Sr after the end of nuclear weapon testings period and the Chernobyl fallout during short and longer time intervals.

Material and methods

Long-lived radionuclides ^{137}Cs and ^{90}Sr have been regularly analysed at STUK since the 1960's in dairy milk and, after 1986, also on farm-specific milk samples in Finnish Lapland. Milk sampling was started in the Kursu dairy in 1963, and it was continued up until the end of 1987. Samples from the dried-milk factory in Rovaniemi were taken during June 1966 – 1975, and dairy milk from the Rovaniemi dairy from 1986 onwards. Farm-specific milk was sampled by the research station of MTT Agrifood Research, Finland at Apukka in Lapland (17 km NE from Rovaniemi) during September 1975 - January 1977 and May 1986 - August 1991, and at Pöykkö farm in Vikajärvi (30 km NE from Rovaniemi) from June 1991 onwards.

The dairy milk represented the whole sampling area of the dairies, and was sampled weekly and combined monthly for ^{137}Cs and ^{90}Sr analysis. Farm-specific milk samples were taken once a month and represented one to three days' milk.

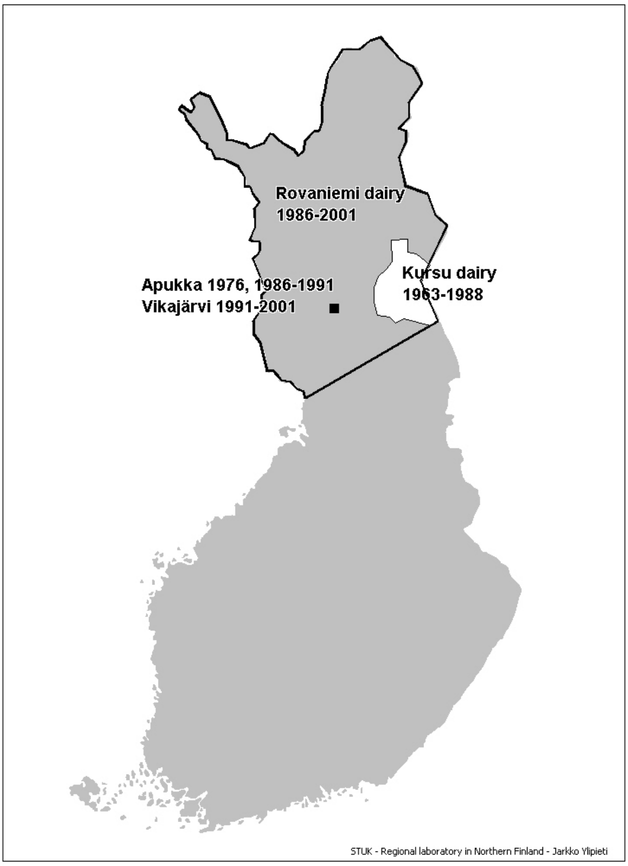

Fig.1. Sampling areas of Kursu and Rovaniemi dairies and sites of farms at Apukka and Vikajärvi.

The dominant soil type in Lapland is peat, and 40% of the cultivated soils are peat soils, 25% till and 35% fine-sand soils. The pastures are poorer, with lower concentrations of potassium and calcium, as well as pH, than in other parts of Finland. There are no clay soils, typical of southern Finland, in the area.

The ecological half-lives for the decrease in ^{137}Cs and ^{90}Sr were estimated using log-linear regression analysis performed on monthly or annual mean values.

Results and discussion

Although total ^{137}Cs deposition from nuclear weapon testings was about the same as the deposition after the Chernobyl fallout, the ^{137}Cs concentrations in milk during the 1960's were considerably higher than those after the Chernobyl accident. The pre-Chernobyl fallout, which was characterized by an annual maximum in summer, resulted in significant direct contamination to growing crops, whereas the Chernobyl deposition occurred before the start of the growing season in Lapland. The direct contamination of growing crops resulted in higher contamination levels compared to the short-term deposition after the Chernobyl accident before the start of the growing season (foodchain hay – cow milk). The time trends of ^{137}Cs and ^{90}Sr concentrations in milk are shown in Figs. 2 and 3, respectively.

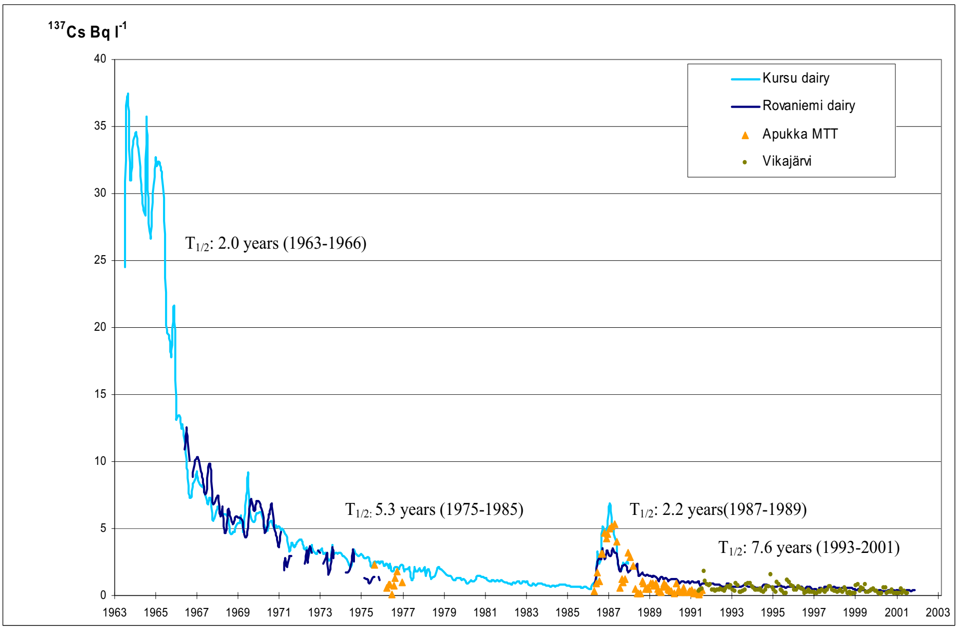

Fig. 2. ^{137}Cs concentrations in milk in Lapland 1963-2001.

The increase in the ^{137}Cs contents in milk due to the Chernobyl fallout was clearly visible in July, and the peak lasted until summer 1987. At the beginning of 1987 the ^{137}Cs concentrations in Kursu dairy milk were twice as high (5 – 7 Bq l^{-1}) as in milk from the Rovaniemi dairy (3.5 Bq l^{-1}). The difference was due to the high frequency of peat soils and to the higher fallout in the area of Kursu dairy compared to the average deposition in Lapland. The peak concentrations in the Apukka farm milk were at about the same level as in Kursu dairy milk, and decreased after summer 1988 to below the level of the Rovaniemi dairy milk. The concentrations of ^{137}Cs in Vikajärvi farm milk were similar to those found in Rovaniemi dairy milk.

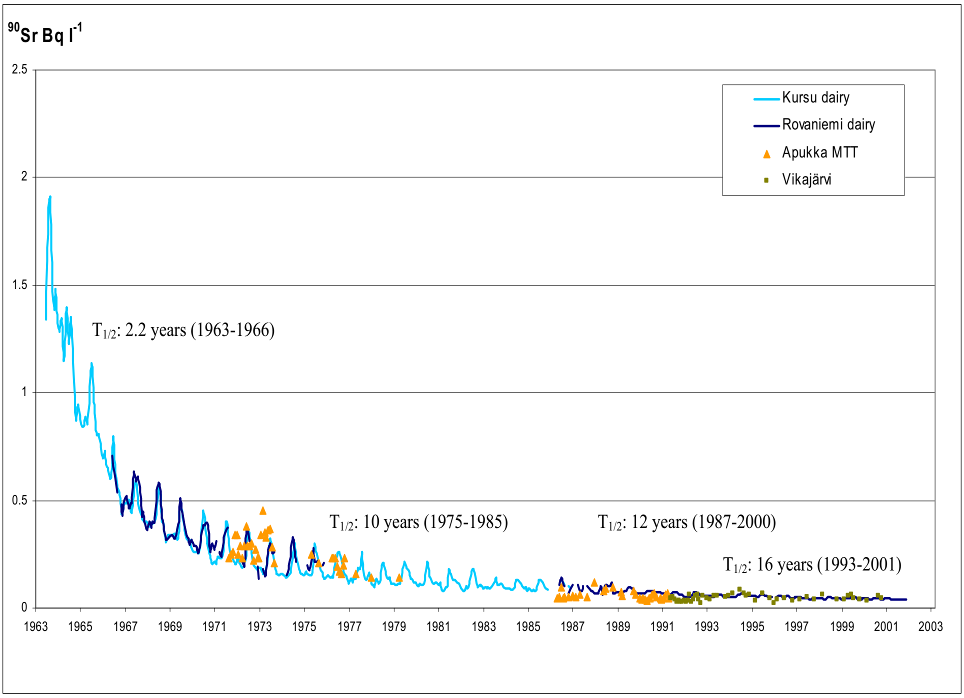

Fig. 3. ^{90}Sr concentrations in milk in Lapland 1963-2001.

^{90}Sr in milk in Lapland mainly originates from fallout from nuclear weapon testings. There was no difference in the ^{90}Sr concentrations in milk from the Kursu and Rovaniemi dairies in July – December 1986 (0.1 Bq l^{-1}), and no increase was noticed in 1987. The concentrations in the Apukka farm milk were slightly lower.

After the nuclear weapon testings period the fallout continued at lower, but not negligible levels for several years. The half-lives of ^{137}Cs and ^{90}Sr in milk after the peak values in 1963 were similar for both radionuclides: 2 years in 1963-1966, and 5 years in 1966-1975. Later on, during 1975-1985, the half-life of ^{137}Cs remained at 5 years, but that of ^{90}Sr increased to 10 years, due to the fact that the long-term bio-availability of ^{90}Sr in the environment is higher than that of ^{137}Cs. The half-lives of ^{137}Cs in milk after the peak concentrations resulting from the Chernobyl accident were similar to the half-lives in the 1960's and 1970's.

The half-lives of ^{90}Sr, were after the first rapid decrease in the 1960's, almost twice as high as those for ^{137}Cs. The half-lives for ^{137}Cs in dairy milk and farm milk were about the same (7-8 years) during the 1990's, as well as those for ^{90}Sr during the 1970's. There were larger fluctuations in the monthly concentrations of ^{137}Cs in farm milk than in dairy milk, but the annual decrease was about the same.

Table 1. The ecological half-lives for the decrease in activity concentrations of ^{137}Cs and ^{90}Sr after the nuclear weapon testings fallout period and the Chernobyl fallout for short and longer time intervals.

Period	$T_{1/2}$ (years), ^{137}Cs		$T_{1/2}$ (years), ^{90}Sr	
	Kursu	Rovaniemi	Kursu	Rovaniemi
1963-1966	2.0		2.2	
1966-1975	4.8	3.2	5.1	6.1
1975-1985	5.3		10	
1987-1989		2.2		14
1989-1993		4.1		6.9
1993-2001		7.6		16

There were annual variations, with maximum concentrations in summertime, for ^{137}Cs and ^{90}Sr in milk originating from southern Finland. The same variation was also be seen for ^{90}Sr in milk produced in Lapland, but not for ^{137}Cs. The reason for this difference in behaviour is so far unknown, but it may depend on differences in soil types, pasture hay species and fertilization practises.

The whole of Lapland is vulnerable to radioactive contamination. The transfer of ^{137}Cs into milk from peat soils is more than two times higher than in the clay soils of southern Finland, and the half-lives observed in milk in Lapland are twice as long as those in intensively-cultivated clayish pastures.

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11j ^{137}Cs concentrations in reindeer meat in the Paistunturi, Ivalo and Kemin Sompio reindeer-grazing co-operatives during 1986-2000

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Introduction

The ^{137}Cs concentration in reindeer meat varies throughout the year due to changes in food selection. During summer, the reindeer eat herbaceous vegetation, and in the autumn large amounts of mushrooms - if available. In winter they prefer to eat ground and arboreal lichen, and these have a much higher radiocaesium concentration than the vegetation eaten during summer. The increase in the reindeer stock in Finland as well as in Norway and Sweden, has lead to overgrazing and degradation of the lichen ranges. The range of winter fodder available for reindeer today is not the same as that in the 60's and 70's following the nuclear weapons fallout period and the previously used ecological half-life values may no longer be applicable to the present situation.

After the accident at Chernobyl in 1986 an extensive program was started to monitor ^{137}Cs concentrations in the reindeer meat produced in Finland. At the present time the program, which is carried out every 5th year, covers the whole of the Finnish reindeer herding area, i.e.56 co-operatives. In addition STUK annually measures samples from the 11 northernmost Sami reindeer herding co-operatives as well as from 3 co-operatives close to the Russian border. In this paper the decrease in the ^{137}Cs concentration in reindeer meat in three co-operatives, Paistunturi, Ivalo and Kemin Sompio, are presented and ecological half-lives estimated. The locations of the co-operatives are shown in Fig. 1.

Materials and methods

The meat samples were provided by the reindeer co-operatives and represent meat from the neck of the animal. Pure meat was separated from the cervical vertebrae and sinew, dried at 105°C, homogenised and measured gammaspectrometrically in 35 ml or 100 ml plastic jars. In Finland the

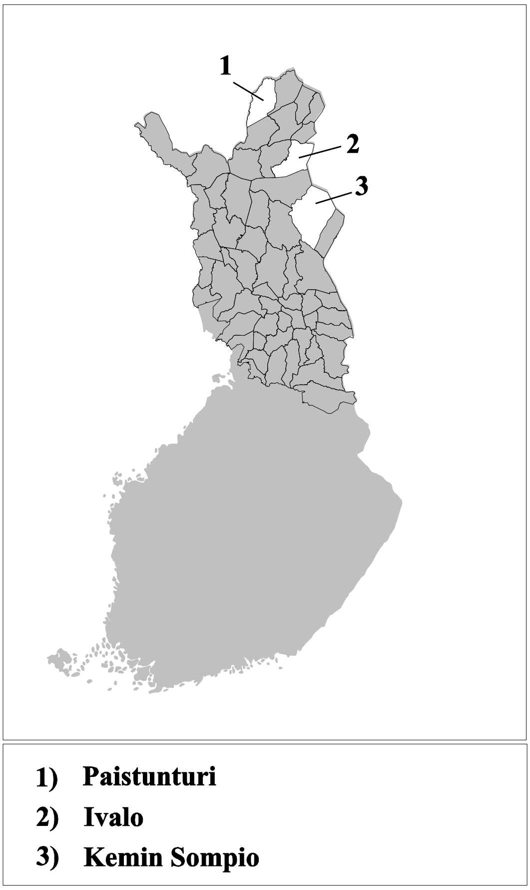

Fig. 1. Location of the Paistunturi, Ivalo and Kemin Sompio reindeer co-operatives on a map showing all the co-operatives in Finland.

reindeer production year starts on June 1st. The slaughtering season is from September/November to February/March, depending on the latitude, temperature and thickness of the snow cover. The annual average winter ^{137}Cs concentrations for the Paistunturi, Ivalo and Kemin Sompio co-operatives presented in this paper represent reindeer slaughtered during the period October 1st to April 30th. The few samples obtained after May 1st and before the end of green vegetation period represent animals feeding on summer fodder. These reindeer were calves or adults killed in traffic or other accidents.

Result and discussion

Although the amount of Chernobyl fallout in Northern Finland was low there were some regional differences. According to lichen sampling in 1986-1987, the ^{137}Cs concentrations[1] were lowest in the northernmost part of the reindeer herding area (the Kaldoaivi co-operative), the Paistunturi area received significantly more, the Ivalo area had low levels, and Kemin Sompio more. In addition to the Chernobyl fallout, the lichen pastures also contained ^{137}Cs from the global fallout period. The unevenness of the Chernobyl fallout was visible in the reindeer meat obtained during the first slaughtering season (1986-1987) after the accident. In Paistunturi and Kemin Sompio the average ^{137}Cs concentration was 1100 Bq/kg, and in Ivalo 600 Bq/kg.

During 2001-2002, 15 years after the accident, the caesium concentrations in Ivalo and Paistunturi reindeer had decreased to the level of 150 Bq/kg, and in Kemin Sompio reindeer to 300 Bq/kg (Fig. 2). The ecological half-lives for the decrease were estimated using loglinear regression analysis. In Ivalo the ^{137}Cs concentrations had decreased steadily with an effective ecological half-life of about 6 years. In Paistunturi there had been a rapid decrease during the first year (half-life about 1 year), and since 1987 a half- life of 6.5 years. In Kemin Sompio the concentrations had decreased during the first three years with a 3-years half-life, and since 1989 with an 8-years half-life.

The trends of the ^{137}Cs concentrations in meat also reflect the condition of winter pasture, because the concentrations in the meat during the slaughtering season correlate with the availability of radiocaesium in the fodder. According to our results the reason for the decrease in ^{137}Cs in the Paistunturi reindeer meat to the same level as in the Ivalo reindeer is due to the decrease in the availability of lichen fodder that occurred soon after the Chernobyl accident. The shortage of ground lichen was also evident during STUK's sample collection trips. In Ivalo there did not seem to be any major changes in winter fodder availability. In the area of the Kemin Sompio co-operative, the lichen pasture situation seems to be still relatively good. It is located in a forest area, in contrast to the rather barren treeless mountain (fjeld) area at Paistunturi. In Ivalo area there are more forest than barren mountains. The pre-Chernobyl concentrations in lichen in Kemin Sompio have also been higher, because one reindeer meat sample from the 1985-1986 slaughtering season contained ^{137}Cs 580 Bq/kg, while the average concentration for the whole reindeer herding area was 300 Bq/kg.

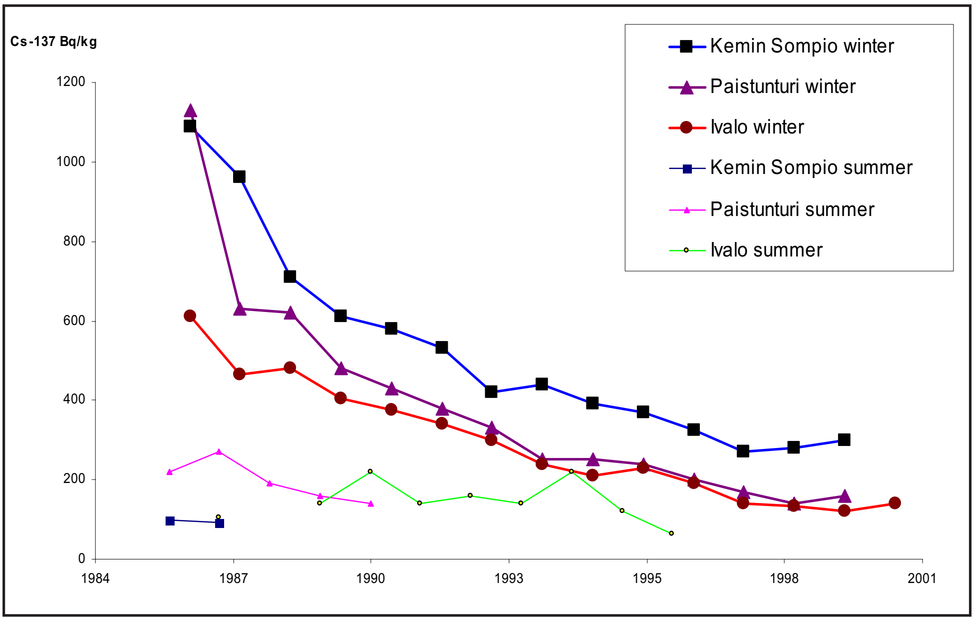

Fig. 2. Time series for annual average ^{137}Cs concentrations in fresh reindeer meat in the Paistunturi, Ivalo and Kemin Sompio reindeer co-operatives after the Chernobyl accident. Winter slaughtering period October 1st - April 30th. The summer concentrations represent a few animals killed in accidents.

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11k Activity of Cs-137 in Forest Mushrooms in Poland in 2001

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Abstract

After the Chernobyl accident systematic measurements of ^{134}Cs and ^{137}Cs activity in forest mushrooms were conducted. The samples, mainly of three kinds of mushrooms, were collected all over Poland. The ratio activity of ^{137}Cs to ^{134}Cs was also estimated. This ratio has been higher than in fallout and other tested foodstuffs. The analysis of mushrooms showed that the activity of caesium is much higher than in other foodstuffs until now. The results of ^{137}Cs activity are presented in mushroom samples collected during season June-October 2001.

Introduction

The Chernobyl accident brought into problem of radioactive contamination of forest mushrooms. The activities of caesium in mushrooms were much higher than in other tested foodstuffs. Such a situation continues nowadays.

Higher contamination of mushrooms results from the specificity of caesium behaviour in forest ecosystem. Caesium slowly penetrates soil and accumulates in litter, lichen and mosses, where the mushroom spawn lives. The durability of this accumulation is confirmed by high concentration of ^{137}Cs in forest mushrooms measured in 1985. The range of results is from a few dozen to above hundred $Bq \cdot kg^{-1}$ fresh mass, while the activity of ^{137}Cs in other tested agriculture products and foodstuffs was below 1 $Bq \cdot kg^{-1}$.

The activity ratio of ^{137}Cs to ^{134}Cs, released during the Chernobyl accident, was 2 to 1.

The same ratio, 2.03±0.02, was registered in total fallout and non-forest foodstuffs in May 1986. This ratio, corrected on 28 April 1986, was 2.60±0.08 for all kinds of forest mushrooms. It can be assumed that about 25%-30% of ^{137}Cs registered in forest mushrooms in 1986 resulted from fallout of nuclear weapon tests [1].

Material and methods

Sampling

Forest mushrooms: Boletus edulis, Cantharellus cibarius and Xerocomus badius, the most popular species of mushrooms in Poland, were picked up in 6 forest regions in Poland in consecutive years to analyse changes of radioactive contamination with time (figure 1). These regions cover the parts of the country where different level of caesium contamination occurred in 1986. About 1500 samples of mushrooms were collected in 2001.

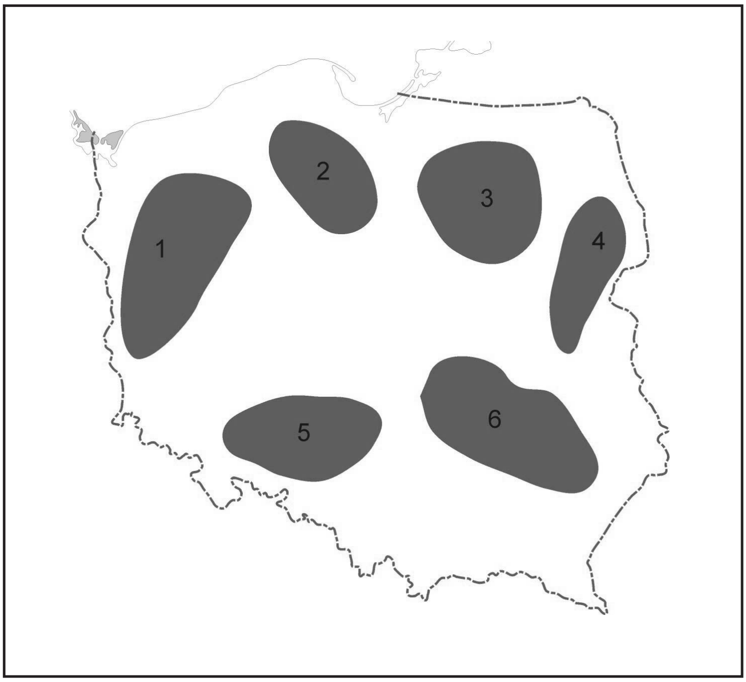

Figure 1. Forest regions in Poland.

Sample preparation and analytical methods

Samples were cleaned from leaves, needles and lichens and homogenised. The caesium activity, in Marinelli geometry, was determined by gamma spectrometry with NaI(Tl) or HPGe detectors. Calibration of the spectrometers was performed using mix-g standard solution. The correctness of calibration was checked in international comparisons organised by Society for the Promotion of Quality Control in Medical and Radiotoxicological Analysis (PROCORAD) and IAEA.

Results and discussion

The mean activities of ^{137}Cs in forest mushrooms were on the level of several hundred Bq per kg fresh mass, in single samples a few thousands Bq·kg^{-1} in 1986. The mean activities increased in 1988-1989. Litter and mosses, contaminated by caesium directly, decomposed and caesium accumulated in them became absorbed by mushroom spawn. It is very difficult to recognise a trend towards decrease of ^{137}Cs activity in forest mushrooms in the following years. The mean and range of activities of ^{137}Cs in mushroom samples in 1986-2001 are presented in table 1. The activity of ^{134}Cs in mushrooms was about 35% of ^{137}Cs activity in 1986 and decreased quickly with radioactive decay to below 1% in 2001 [2,3].

Considerable difference of radioactive contamination of different kinds of forest mushrooms, sampled from the same forest area in the same time, was observed. This is caused by different properties of absorption and mechanism of accumulation by individual kinds of mushrooms.

Table 1. Activity of ^{137}Cs in forest mushrooms in Poland in 1986-2001 *(Bq·kg^{-1} fresh mass).*

	Boletus edulis ***(Bqkg^{-1})***		**Cantharellus cibarius** ***(Bqkg^{-1})***		**Xerocomus badius** ***(Bqkg^{-1})***	
	mean	**range**	**mean**	**range**	**mean**	**range**
1986	36	12-135	67	8-511	380	60-1547
1987	38	14- 94	64	9-612	377	87-2312
1988	90	38-233	46	13-129	634	57-1730
1989	62	7-164	37	16-117	669	89-2100
1990	75	11-217	77	18-351	551	36-1909
1991	69	14-224	135	8-750	509	134-1284
1992	102	23-610	49	11-196	525	90-1959
1993	89	8-427	50	5-300	371	19-1190
1994	37	5-66	27	5-132	363	43-838
1995	48	10-103	46	9-172	373	156-1017
1996	92	14-433	31	5 - 77	197	47-453
1997	60	18-138	20	<5 - 61	462	59-1058
1998	80	7-254	48	<5-267	373	63-1863
1999	67	<5-1002	61	<5-603	261	17-1191
2000	59	<5-678	50	<5-557	166*	20-662*
2001	61	<5 - 401	47	<5 - 294	172**	6 - 578**

* excluding one sample of 3146 Bq·kg^{-1}
** excluding one sample of 1215 Bq·kg^{-1}

In the first years after the Chernobyl accident, an attempt could be made to compare the mean concentration of ^{137}Cs in forest mushrooms taken from different regions in Poland to ^{137}Cs activity in total fallout in these regions in 1986 [4,5]. But high dispersion of activity of individual samples taken from the same region and the same forest was observed. Local differences of radioactive contamination of the lowest forest part were caused by e.g. keeping of caesium isotopes by tree-tops. The falling leaves and needles increase the level of caesium in litter as a consequence.

The results show that ^{137}Cs activites are still on a high level in mushrooms. The mean activities in Boletus edulis and Cantharellus cibarius samples are on the same level as in previous years. A decrease of activity in Xerocomus badius samples can be observed, but it may be connected with small number of samples of this species.

The activity of ^{137}Cs in mushrooms from sampling regions in Poland in 2001 is shown in table 2.

Table 2. Activity of ^{137}Cs in mushrooms in forest regions in Poland in 2001 *(Bq·kg^{-1} fresh mass)*

Region	Boletus edulis			Cantharellus cibarius			Xerocomus badius		
	N	Mean	Range	N	Mean	Range	N	Mean	Range
1	337	63	<5-401	340	40	<5-159	80	144	6-259
2	82	61	14-178	97	55	5-201	5	263	170-498
3	64	57	<5-262	167	56	<5-149	7	225	32-455
4	36	46	6-133	109	43	<5-212	11	319*	21-578*
5	9	71	9-156	10	96	16-195	3	141	90-177
6	23	58	16-96	105	51	<5-294	4	183	150-220

* excluding one sample of 1215 Bq·kg^{-1}

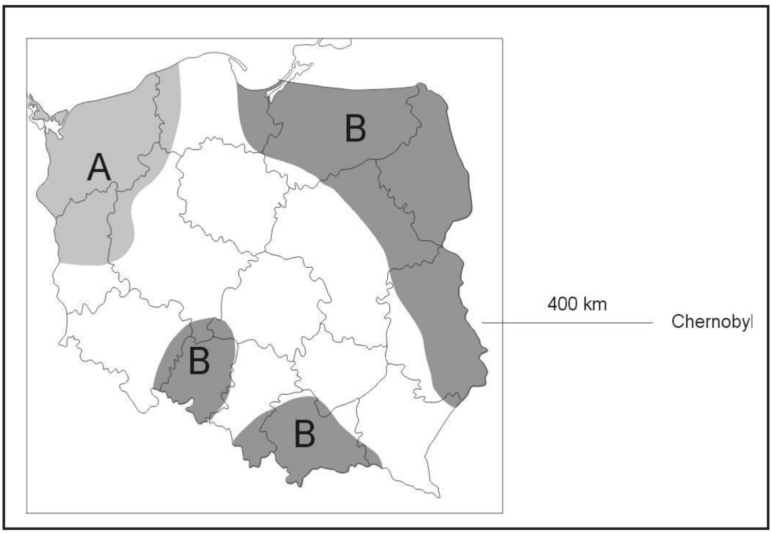

Figure 2. The regions of different level of contamination in 1986 (A-lower level, B- higher level).

In 2001 there was no relationship between the mean activities from particular regions and fallout in these regions in 1986. Figure 2 shows the irregularity of radioactive contamination of total fallout after the Chernobyl accident. The forest region 1 is situated in low level area of contamination (A), regions: 3,4,5 – in higher level (B), regions: 2,6 – in intermediate level. But a wide range of results in particular samples of each mushrooms species in each region is observed. It testifies that clear local differences of ^{137}Cs contamination in forest litter do not change.

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11m Effective dose and time-integrated effective dose to humans from internal contamination of ^{134}Cs and ^{137}Cs: Results from a compilation of a Swedish national database of internal body burden of radiocaesium in various populations between 1964 and 2002

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Abstract

A compilation of data on the whole-body burden of ^{134}Cs, ^{137}Cs och ^{40}K in various Swedish populations between 1964 and 2002 has been made. The compilation was carried out in co-operation with the Department of Radiation Physics in Malmö, the Swedish Radiation Protection Authority (SSI), the Swedish Defence Research Agency Department (FOI) and the Department of Radiation Physics, Göteborg University. Individual body burden values have been inserted into a calculation spread sheet, with data on body weight, gender, age, occupation (if available) and place of residence. The database enables a study of the time-pattern and geographical dependence of radiocaesium transfer from ground deposition to man, and the associated absorbed dose. The Swedish government has assigned SSI the responsibility for obtaining and assuring one of Sweden's national environmental quality objectives, *A Safe Radiation Environment*. A natural consequence of this responsibility is that a means for quantifying progress towards the objective is necessary. The data compilation discussed here is one important component in the national environmental monitoring programme that is currently (2002) being developed at SSI. The

new program for monitoring environmental radiation has the goal of following geographical and ecological differences in the radiological effects to both mankind and the environment, and assessing relevant doses and risks.

During the period between 1964 and 2002, fallout of radiocaesium from nuclear weapons tests (only ^{137}Cs) and from the Chernobyl accident in 1986 (both ^{134}Cs and ^{137}Cs) have occurred in Sweden. The ground deposition of the radiocaesium has gradually been transferred through different ecological pathways to man. From the database it can be deduced that large regional variations in the body burden of radiocaesium in man have occurred through this period. Three populations exhibit considerably higher body content levels than others; *i.)* reindeer herders in central Sweden (both pre- and post-Chernobyl fallout in 1986), *ii.)* hunters in the region of highest Chernobyl fallout, and *iii.)* urban and rural populations outside the city of Gävle (Gävle was the most affected community in Sweden by the Chernobyl fallout). The estimated time-integrated effective dose during a period of 50 years from the Chernobyl fallout is about 10 mSv for reindeer herders, which is, on average, 10 to 100 times higher than urban populations in the three major Swedish metropolitan areas (Malmö, Göteborg and Stockholm). The main reason behind these differences is a combination of two factors; *i.)* large regional variations in the ground deposition of radiocaesium and *ii.)* large variations in the dietary habits of the different investigated populations.

Introduction

The Swedish government has assigned SSI the responsibility for obtaining and assuring one of the fifteen national environmental quality objectives, *A Safe Radiation Environment*(1). One of the criteria for a safe radiation environment is that no human receives dose contributions from the sum of all man-made sources higher than 1 mSv per year. It is therefore important to compile existing experimental data on internal contamination levels in man, and relate these values to the levels of fallout and dispersion of radioactive contaminants in the environment. During spring 2001 a project was launched, with the SSI as the co-ordinator, where data records on body burdens of radioactive caesium in various Swedish populations, studied during the past four decades by different Swedish organisations, were compiled into a single database for a general assessment (Table 1). The aim of the assessment was to study the effects in terms of radiation dose and late effects from different exposure pathways and identify which of these pathways could be of most importance in a future exposure scenario. The database is also intended to be integrated into a general database that includes contamination levels of man-made radionuclides in various species, and used as reference data for the environmental monitoring programme at SSI.

The global fallout of fission products such as ^{90}Sr, ^{131}I, and ^{137}Cs from the atmospheric nuclear weapons tests during the 1950's and 60's initiated a number of studies on the body burden of radiocaesium in a number of human populations in Sweden(2-6). In a study from 1960 it was revealed that reindeer herders had significantly higher body burden levels of ^{137}Cs compared with what was found in urban areas in Sweden(2). During the following decades several studies on the body burden of ^{137}Cs in reindeer herders in the Nordic countries were carried out as well as in other Arctic populations(7,8). Certain ecosystems were then demonstrated to be more sensitive to transfer of radioactive deposition to man, especially within the Sub-Arctic ecosystems that include a minimum of modern day agriculture and forestry(9). Individuals who consumed reindeer meat or game as well as forest products such as mushrooms and wild berries were a special group at risk. At the same time long-term studies on the caesium body burden in humans living in large urban areas were carried out as a reference for the more critical populations investigated(3,6). After the Partial Test Ban Treaty was signed by the superpowers in 1963, the number of atmospheric detonations decreased considerably, leading also to a gradually decreasing global fallout.

The studies began again due to the Chernobyl accident in 1986, which resulted in a large fallout of ^{134}Cs and ^{137}Cs in Sweden. This time the deposition was much more inhomogeneously distributed, where certain communities obtained more than 100 kBq/m^2, compared with 2-3 kBq/m^2 that was deposited from the earlier global fallout from atmospheric detonations(10-11). Apart from the reindeer herders, which had already been identified as a critical group in terms of potential transfer of radioceasium, large studies were then also conducted on other populations at risk in Sweden, such as hunters and farmers living in regions with a high deposition of radiocaesium(5). These categories were found to have higher body burdens of ^{134}Cs and ^{137}Cs, but in general the contamination levels in the Swedish populations were lower in relation to the ground deposition, when compared with the 1960's(12).

Materials and methods

Data on individual body burdens in different study groups that were conducted by the Swedish Defence Research Agency Department (FOI), the Swedish Radiation Protection Authority, as well as two university departments;

Dept. of Radiation Physics in Malmö (Lund university) and in Göteborg (Göteborg university), were compiled to one single database. Prior to the data compilation the reporting format and the parameters to be considered were agreed upon by the different participating departments. In addition to the whole-body contents of ^{137}Cs, ^{134}Cs and ^{40}K (an important physico-chemical analogue to caesium, omnipresent in most living organisms), values on the investigated individuals age, sex, body weight, lieu of residence and main occupation or profession (if available) were included in the database. The latter parameter was used to categorise the study of populations in the following categories: *i.)* farmers, *ii.)* hunters (leisure time or professional) *iii.)* reindeer herders, *iv.)* fishermen or *v.*) urban populations. Table 1 gives an overview of the studied populations and the departments in charge of the studies.

Table 1.

The Swedish Defence Research Agency (FOI)	Reindeer herders in Västerbotten (FOI Sami)	1988 – 2001
	Hunters in central Sweden (FOI Hunters)	1994 – 1998
	Reference group in Lapland (FOI Reference)	1991 – 1996
The Swedish Radiation Protection Authority (SSI)	Reference group: Employees at SSI (Stockholm)	1966 – 2001
	National survey of subjects (randomly selected subjects in Sweden; stratified for the regional variation in ^{137}Cs-deposition)	1987 and 1994
	Rural inhabitants outside the city of Gävle	1986 – 1998
	Urban inhabitants in the city of Gävle	1986 – 1998
Dept. of Radiation Physics in Göteborg	Reference group: Employees at Göteborg (Göteborg)	1986 – 1989
Depts. of Radiation Physics in Malmö and in Lund	Reference group: Employees at Dept. of Radiation Physics in Malmö (urban population living in Malmö)	1986 – 1994, 2002
	Reference group: Employees at a factory in Lund (Skåne)	1964 - 1994
	Reindeer herders in Härjedalen (bordering to Norway)	1965 - 1976
	Inhabitants living outside the Barsebäck NPP	1974 - 1975

The concentration of ^{137}Cs and ^{134}Cs as a function of time could be determined from the database. The effective dose rate (mSv y^{-1}) was estimated by using the relationship between body burden of ^{134}Cs and ^{137}Cs and effective dose described by Leggett *et al.* in 1984[13]. Curve fitting of exponential expressions describing the gradual decrease of the human concentrations of radiocaesium with time was employed enabling time-integration of the effective dose during a 50-year period. Using annual deposition data from 1950-1990 from Denmark(14), a country neighbouring to Sweden, a coarse estimation of the time-integrated effective dose to the populations investigated in the 1960's could be obtained for the period 1945 – 1995.

Results

In Fig. 1 the annual effective dose from ^{137}Cs and ^{134}Cs to the various Swedish populations is plotted as a function of time. Swedish reindeer herders exhibit the highest values both before and after the Chernobyl accident in 1986 with peak dose rates at 1–2 mSv y^{-1}. In Table 2 is given the estimated individual time integrated effective dose during a 50-year period due to nuclear weapons and Chernobyl fallout respectively.

Table 2. Mean individual time-integrated effective doses in mSv calculated from the average ^{137}Cs-concntertaions in different Swedish populations.

Population	Global Fallout ^{137}Cs (1945-1995)	Chernobyl ^{137}Cs (1986-2036)	Chernobyl ^{134}Cs (1986-2036)
FOI Sami (Most Affected Areas)	N/A	11	2.4
FOI Hunters [a]	N/A	1.2	N/A
FOI Reference	N/A	0.6	N/A
SSI Control group	0.27	0.1	0.03
Gävle inhabitants (urban)	N/A	0.5	0.13
Gävle inhabitants (rural)	N/A	1.2	0.37
Göteborg	N/A	0.034	0.02
Skåne (Malmö group)	N/A	0.08	0.02
Skåne (Lund Reference Group)	0.22	0.033	0.007
Härjedalen Sami	27	N/A	N/A
National survey	N/A	0.13	0.042

[a] Average over three different hunting teams at different locations in Sweden.

Conclusions

The data show that of the Swedish populations studied the reindeer herders have obtained, on average, the largest individual effective doses from internal radiocaesium contamination. The time integrated effective dose to reindeer herders in Härjedalen is estimated at 27 mSv from nuclear weapons fallout (1945-1995) and for reindeer herders in the most affected areas of the Chernobyl fallout an additional 10 mSv is estimated to be received during the period 1986 – 2036. Furthermore, hunters and farmers exhibit about 10 times

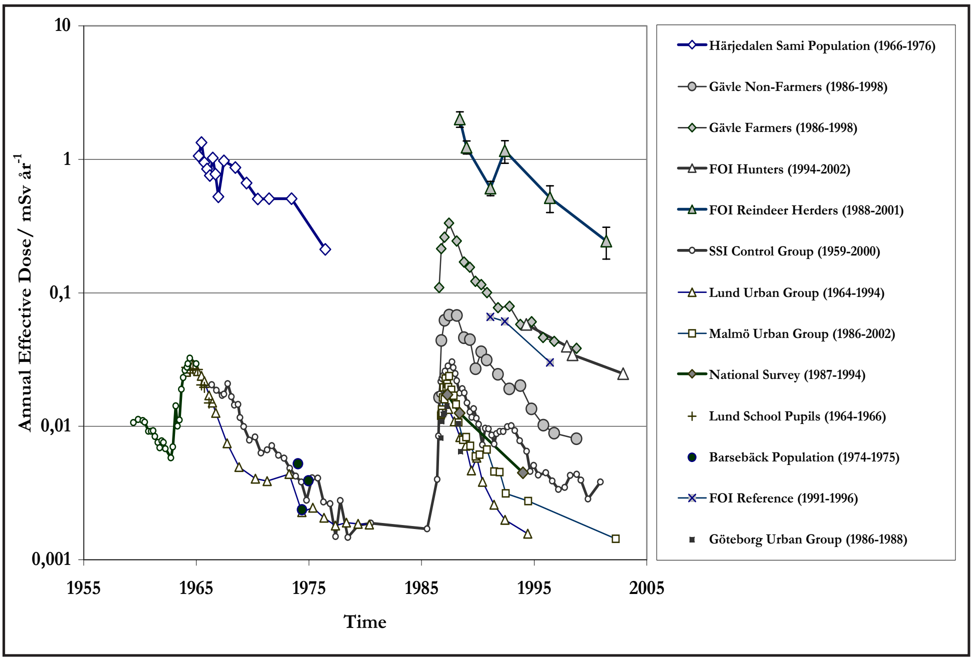

Fig. 1. Effective dose per year from ^{134}Cs and ^{137}Cs to various Swedish populations.

higher effective doses from the Chernobyl fallout than urban residents in the Swedish metropolitan areas (Stockholm, Göteborg and Malmö).

The national average values of the time-integrated effective dose for an individual is estimated to be between 1.3 – 1.6 mSv in Sweden, Finland and Norway according to food consumption data compiled by Aarkrog in 1994(7). This can be compared with 0.13 mSv (^{137}Cs only) estimated from the national survey conducted in Sweden. However, using whole-body counting data from control groups in Helsinki (Finland) and Sel (Norway) from the years 1986 to 1990(7), indicates that the difference between these countries is somewhat less; with the Swedish values being a factor of 5 times lower.

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11n Assessment of radionuclide content in the human body. The influence of inhomogeneously distributed activity.

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Abstract
In a fixed mode whole-body measurement, a single fixed position of the detector relative to the person to be measured is used. A fixed mode measurement does not give information about the distribution of activity in the body. Consequently, the same amount of activity in the body can result in different detector responses depending on the activity distribution. The accuracy of an assessment of the total body content therefore depends on how well the activity distribution, used in the calibration of the whole-body counter, resembles the actual distribution in the body. The Risø whole-body counter is a fixed mode counter. To investigate the influence of different activity distributions on the counting efficiency a detailed calibration of the Risø whole-body counter has been made. The calibration was made with an anthropomorphic brick phantom. A series of measurements were made to determine the counting efficiencies for ^{60}Co and ^{137}Cs activity placed in single bricks at different positions in the phantom. The counting efficiency for an arbitrary activity distribution can be calculated from proper weighting and summing of the measured efficiencies for activity in the single bricks.

Introduction
Whole-body counting of body contents of radionuclides is primarily done in two ways, the scanning mode and the fixed mode. In scanning mode measurements are made while the detector is moved along the person. This is usually done with a collimated detector along the centre axis of the person and information on the activity distribution is obtained directly. In fixed mode measurements a single fixed position of the detector relative to the person is used, and thus measurements will not give information about the activity distribution in the body. Consequently, the same amount of activity in a body can result in different detector responses depending on the distribution of activity. This is

predominantly due to the variation in counting efficiency with distance to the detector and due to the shielding effect of body tissue. The accuracy of an assessment of the total body content therefore depends on how well the activity distribution used in the calibration of the whole-body counter resembles the actual distribution in the body.

The Risø whole-body counter is a fixed mode counter. In this counter the person is sitting in a chair with the front facing the detector (Ge-detector). Most points on the front side will in this position have almost the same distance to the detector, and thus the geometry to a certain extent compensates for the lack of knowledge about the activity distribution. Usually a homogeneously activity distribution is assumed when assessing the body content although most radionuclides are distributed unevenly in the body.

To investigate the influence of different activity distributions on the counting efficiency a detailed calibration of the Risø whole-body counter has been made at 662 keV (^{137}Cs), 1173 keV (^{60}Co), and 1332 keV (^{60}Co) and a less detailed calibration at 122 keV (^{57}Co). From these calibrations, ratios of apparent activity (homogeneous distribution assumed) to true activity have been calculated for a few activity distributions and energies.

Measurements

Measurements were made with the anthropomorphic phantom “IRINA”, developed by the “Research Institute for Industrial and Sea Hygiene” in St. Petersburg, Russia. The phantom is made from tissue-equivalent bricks with mass 0.88 kg (whole brick) and 0.40 kg (half brick) respectively. Two rods with activity can be placed in each brick. The phantom is provided with four sets of rods, namely sets with ^{57}Co, ^{60}Co, ^{134}Cs (not used), and ^{137}Cs.

The phantom configuration (P4) used in the calibrations is build from sixty-nine whole bricks and two half bricks. The head and neck are made of 4 whole and 2 half bricks, the chest of twenty whole bricks, the arms of 8 whole bricks, the abdomen of 13 whole bricks, the thighs of 14 whole bricks, and the legs of 10 whole bricks. The phantom is shown in Figure 1. The labelling letters refer to head, breast, arms, stomach, thighs and legs respectively.

In two measurements series (^{60}Co and ^{137}Cs) rods were placed successively in each brick of the phantom (except for a few bricks) and the counting rates were measured. Bricks placed symmetrically (seen from the detector) for example T51 and T57, were measured at the same time (two rods in both bricks) in order to save time. The counting rate for one brick was afterwards set to be half the measured rate for both bricks. This symmetry

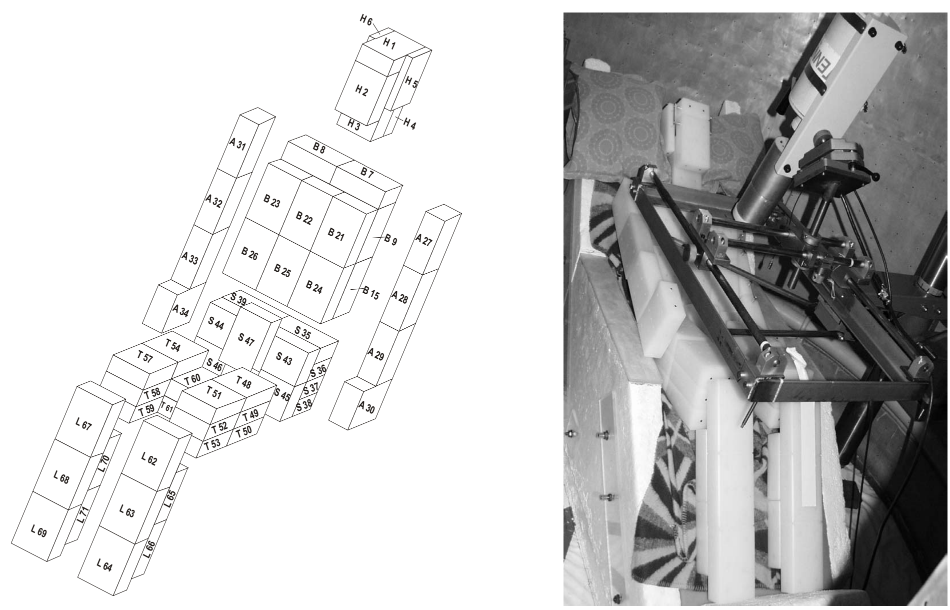

Figure 1. The "IRINA" phantom (configuration P4) used in the measurements. The phantom represents a person of height 170.5 cm and with a weight of 61.5 kg. The labelling letters on the left drawing refer to head, breast, arms, stomach, thighs and legs respectively. The phantom was measured in"sitting geometry" as shown on the photo.

assumption was confirmed from measurements on B9 and B14 (at the back).

Three counting rate measurements were made with ^{57}Co, one with rods in S43-S44, one with rods in T50-T56 and one with rods in all the bricks of the phantom.

Results

The maximum counting rate (pr. activity unit) at 662 keV was measured when the ^{137}Cs activity was placed in S47, whereas the maximum counting rate at 1173 keV and at 1332 keV was measured when the ^{60}Co activity was placed in B22. Figure 2 presents the measured counting rates for activity placed in the different bricks expressed as a percentage of the maximum counting rate measured at the given energy. The calculated relative counting rates are equal to relative counting efficiencies.

From the measurements with ^{57}Co activity the ratio of the counting rate for activity in S43-S44 to the counting rate for activity in T50-T56 was calculated to be 20 ± 3. The ratio reflects that T50 and T56 are more shielded (towards the detector) than S43 and S44.

When the relative counting efficiencies, r_i, are known for all the bricks in a phantom, as was the case for 662 keV, 1173 keV, and 1332 keV (the few bricks in the head region that were not measured were assigned efficiencies based on measured efficiencies of the other head bricks) the ratio, R, of the apparent activity - when a homogeneous distribution is assumed - to the true activity, can be calculated for an arbitrary distribution as:

$$R = \frac{\sum_i f_i \cdot r_i}{\sum_i r_i} \cdot N \qquad (1)$$

where f_i is the fraction of the total body activity that is situated in brick i, and N is the total number of bricks.

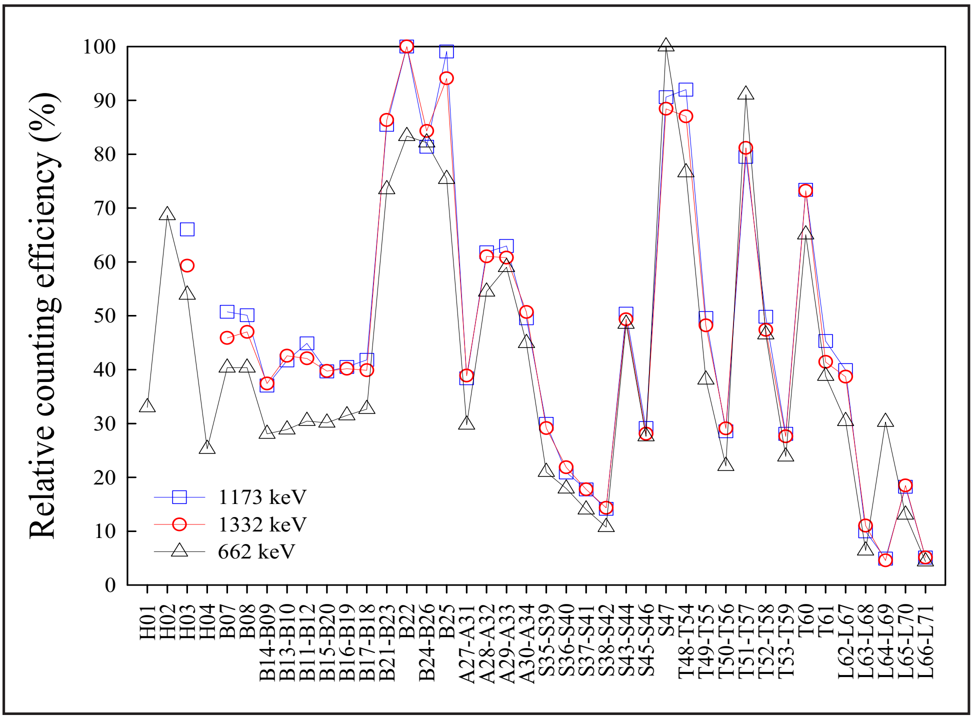

Figure 2. Relative counting efficiencies (measured counting rates for activity placed in the different bricks expressed as a percentage of the maximum counting rate measured at the given energy) for 662 keV, 1173 keV, and 1332 keV photon. For two-label entries on the x-axis, the shown efficiency applies to each of the two bricks.

Table 1. Ratio of apparent activity of ^{60}Co (1173 keV and 1332 keV), and ^{137}Cs (662 keV) in the body (homogeneous distribution assumed) to true activity calculated from equation (1) for four activity distributions.

γ-energy	All activity in the lungs	All activity in the stomach	All activity in the liver	4/5 of activity in the liver, 1/5 uniform in the body
662 keV (^{137}Cs)	0.75 ± 0.02	0.88 ± 0.02	1.24 ± 0.02	1.19 ± 0.02
1173 keV (^{60}Co)	0.94 ± 0.02	0.87 ± 0.02	1.18 ± 0.02	1.15 ± 0.02
1332 keV (^{60}Co)	0.92 ± 0.02	0.87 ± 0.02	1.22 ± 0.02	1.15 ± 0.01

R-ratios were calculated from equation (1) for 662 keV, 1173 keV, and 1332 keV photons for four inhomogeneous distributions of activity. The distributions were: 1) all activity in the lungs, 2) all activity in the stomach, 3) all activity in the liver, 4) 4/5 of the activity in the liver and 1/5 of the activity distributed homogeneously in the body. The organs were formed by three or four elements: lungs (B10, B11, B12, B13), stomach (S43, S44, S35, S39), and liver (B16, B17, B24).

Table 1 presents the calculated ratios for the four distributions. Uncertainties are one standard deviation and arise from counting statistics on the primary data. An assumption of a homogeneous distribution will underestimate the activity when $R<1$ and overestimate the activity when $R>1$.

Equation (1) could not be used to calculate R-ratios from the ^{57}Co measurements; hence an approximate method was used. This method envisages a given distribution of activity in the phantom as a linear combination of the two types of bricks known for ^{57}Co, namely a shielded brick (T50-T56) and a shallow lying brick (S43-S44). From the number of shielded and shallow bricks, N_d and N_s, and the measured counting rates from the shielded brick, the shallow brick and the phantom with all bricks filled with activity (r_d, r_s, and r_{tot}), R-ratios are calculated as:

$$R = \frac{\frac{N_d}{(N_d + N_s)} \cdot r_d + \frac{N_s}{(N_d + N_s)} \cdot r_s}{r_{tot}} \cdot N \tag{2}$$

The linear combination is independent of energy and thus the coefficients N_d and N_s will be the same for any radionuclide (energy), therefore the coefficients can be obtained from the measurements with ^{137}Cs. For a given distribution the R-value calculated from equation (1) for ^{137}Cs is inserted into equation (2). Solving this equation together with the constrain, that the total number of bricks in the distribution (e.g. four in the case of all activity in the lungs) equals $N_d + N_s$, gives N_d and N_s.

Table 2. Ratio of apparent activity of ^{57}Co (122 keV) and ^{60}Co (1173 keV) in the body (homogeneous distribution assumed) to true activity calculated from equation (2) for four activity distributions.

γ-energy	All activity in the lungs $N_d = 2.9$ $N_s = 1.1$	All activity in the stomach $N_d = 2.1$ $N_s = 1.9$	All activity in the liver $N_d = 0.0$ $N_s = 3.0$	4/5 of activity in the liver, 1/5 uniform in the body $N_d = 0.2$ $N_s = 3.5$
122 keV (^{57}Co)	0.39 ± 0.08	0.60 ± 0.08	1.21 ± 0.10	1.13 ± 0.09
1173 keV (^{60}Co)	0.81 ± 0.09	0.91 ± 0.08	1.18 ± 0.10	1.14 ± 0.08

R-ratios at 122 keV (^{57}Co) were calculated from equation (2) for the four distributions mentioned above. The feasibility of the method was tested by calculating R-ratios for 1173 keV photons for the four distributions. These results are shown in Table 2 together with the N_d and N_s values for the four distributions. Comparing the 1173 keV values in Table 1 and Table 2 gives some confidence in the method used. Uncertainties are one standard deviation and arise from counting statistics on the primary data.

Conclusion

From Figure 2 it is seen that the counting efficiency varies as a function of where the activity is placed in the body. As expected positions away from the detector (legs and head) and positions well shielded (e.g. S38) have lower counting efficiencies compared to unshielded positions close to the detector (B22 and S47). The variation of the relative counting efficiencies is almost the same at 662 keV, 1173 keV, and 1332 keV.

From Table 1 it is seen that activity located solely in the lungs or in the stomach will be underestimated if a homogeneous distribution is assumed and counting is made at 662 keV, 1173 keV, or 1332 keV, whereas activity located mainly in the liver will be overestimated. The deviations are within ±25 % of the true value; a deviation figure also considered valid at all energies between 662 keV and 1332 keV. Such deviations will in most cases give minor contributions to the overall uncertainty when body contents are used to calculate internal doses.

At 122 keV (see Table 2) activity located solely in the lungs or in the stomach will result in a larger underestimation of activity compared to counting at 662 keV, 1173 keV, or 1332 keV, whereas activity located mainly in the liver will be overestimated to the same extent.