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Van 14 tot 19 mei 2000 vond te Hiroshima het tiende congres van de IRPA (International Radiation Protection Association) plaats.
Verschillende landgenoten hebben op dit congres een poster voorgesteld. Het Bureau van de Belgische Vereniging voor Stralingsbescherming heeft besloten om de uitgewerkte teksten van deze posters te bundelen in dit nummer van haar Annalen.

Annales de l'Association belge de Radioprotection, Vol. 25, n° 3, 2000

Du 14 au 19 mai 2000 a eu lieu à Hiroshima le dixième congrès de l'IRPA (International Radiation Protection Association).
Plusieurs Belges y ont présenté un poster. Le Bureau de l'Association belge de Radioprotection a décidé de grouper les textes élaborés de ces posters dans le présent numéro de ses Annales.

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Annales de l'Association Belge de Radioprotection, Vol 25, n°3, 2000 pp 193-202 P-4a-225

Radium Contamination Of The Banks Of A Small River Receiving The Liquid Effluents Of A Large Phosphate Plant

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ABSTRACT

A chemical plant in the north of Belgium processes since 1929 large quantities of marine phosphate ore, with a ^{226}Ra content of 1500 Bq/kg, into products suited for animal feeding, using hydrochloric acid (HCl) as dissolution agent instead of the more common sulphuric acid (H_2SO_4). Until 1991, two thirds of the radium was released with the waste water into two small rivers with a concentration of about 20 Bq/l and a flow rate of 1500 m³/h. Adding barium salts has recently reduced the radium concentration of the waste water to 3 Bq/l. The purpose of this study was to map the historical radium contamination of one of the receiving rivers, the Laak, over a distance of 20 km. Enhanced dose rates are found everywhere along the Laak between the discharge points and the merging with the Grote Nete. The contamination is mostly confined to a narrow strip of 5 to 10 m wide on one or on both sides, caused by the periodic dredging of the sediment. The measured dose rates vary between the low natural background of the region, 50 to 80 nSv/h, and 1000 to 2000 nSv/h. The total surface area contaminated above 100 nSv/h amounts to 22 ha. The radiological impact on the surrounding population was estimated. Realistic scenarios for critical groups for the *external exposure* result in doses of a few hundreds of µSv/year. Pastures and maize for animal feeding are currently the only cultures along the contaminated banks of the Laak. This additional step in the food chain reduces the *internal doses* to a few tens of µSv/year. The *inhalation of radon decay products* in open air poses no problem. The construction of dwellings on the contaminated banks would undo this favorable situation as radon gas could accumulate in the dwellings resulting in doses that could exceed the limit for radiation workers.

OVERVIEW OF THE RADIUM DISCHARGES IN LAAK AND WINTERBEEK

For over half a century phosphate ore has been processed in Belgium to produce dicalcium phosphate for use in cattle food. The phosphate ore is from Moroccan origin and has a ^{226}Ra content of 1500 Bq/kg, about two orders of magnitude higher than the world average of the earth's crust of 40 Bq/kg (1). Acidulation of the phosphate rock with HCl results in a waste sludge of mainly CaF_2 retaining about a third of the radium. The remaining two thirds were discharged with the waste water into two small rivers: the Laak and the Winterbeek, with a flow rate of about 1500 m³/h. The waste water also contains high concentrations of salts, mainly $CaCl_2$, and some heavy metals from the raw phosphate ore.

The evolution of the radium concentration of the wastewater is shown in figure 1. Barium salts are added to the production process since 1991 decreasing the radium concentration of the wastewater from 20 Bq/l to 3 Bq/l in 1996 (2). The yearly radium emission into the small rivers is presented in figure 2. The total activity decreased from 280 GBq/year prior to 1991 to 26 GBq/year in 1996 (2). Downstream the discharge points, the banks and the riverbed of the Laak and Winterbeek are contaminated with radium-226. According to an extensive study in the beginning of the sixties (3) the contamination is measurable up to a distance of 50 km.

The river Laak is located in the northeastern part of Belgium at about 20 km from the Nuclear Research Centre of Mol, SCK. It is a small river that slowly flows through a region that is characterized by a predominantly sandy soil. The river drains a surface area of approximately 125 km² and has an annual average flow of 1.2 m³/s with seasonal variations between 0.8 and 2.3 m³/s. On 30 January 1995 a maximum flow of 5.2 m³/s was registered. At ten-yearly intervals, the river was dredged and the sludge was simply deposited on the river banks, except the last time when the sludge was removed and deposited at the industrial repository of the phosphate plant. A few flooding zones occur near the river mouth over a distance of 5 km. These zones are flooded in case of persistent heavy rainfall, which can happen once or a few times a year.

This study was done on behalf of the Flemish Government (AMINAL). The purpose was to chart the radium contamination of the river banks of the Laak over a distance of 20 km from the discharge points to the merging with the river the Grote Nete (2).

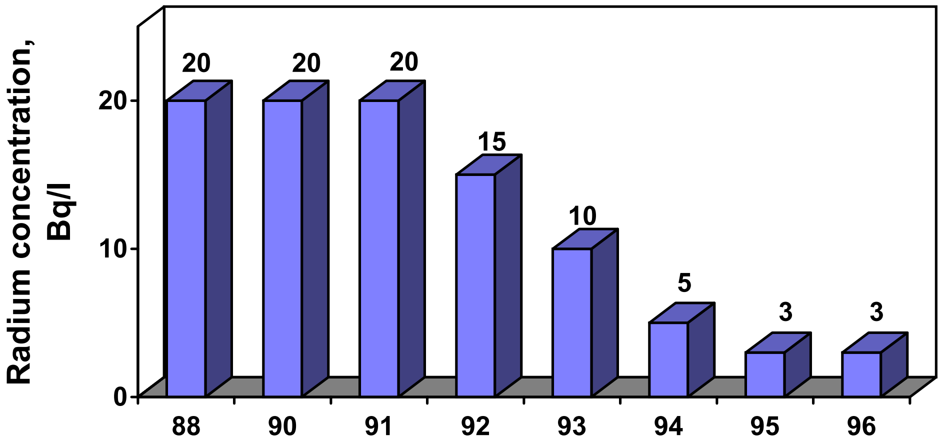

Figure 1. The yearly average radium concentration of the waste water discharged into the Laak and Winterbeek, in Bq/l, from 1988 until 1996.

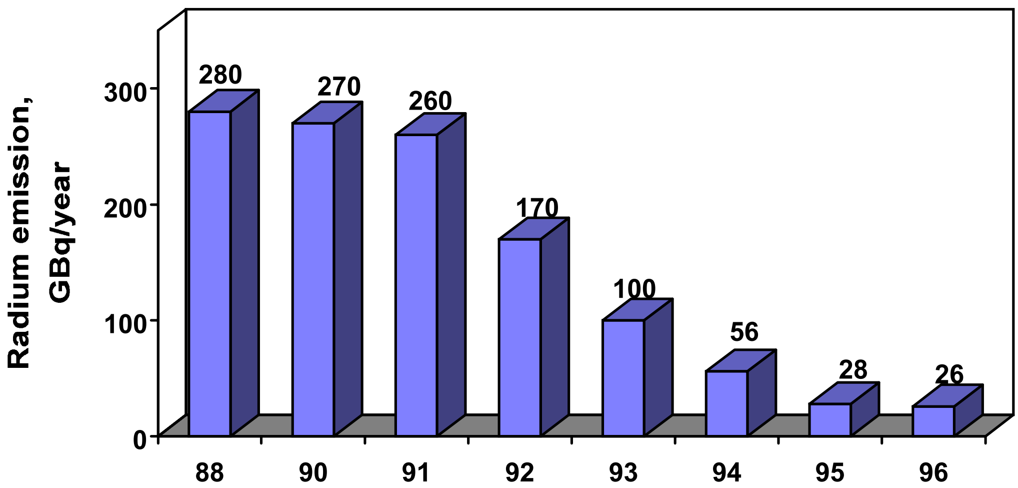

Figure 2. The amount of radium discharged yearly into the Laak and Winterbeek, in GBq/year, from 1988 until 1996.

EXPOSURE TO RADIUM AND RADIUM DECAY PRODUCTS

Radium-226 is a member of the natural uranium series, represented in diagram form in figure 3. It is omnipresent in the environment because of the 4.5 billion years half-life of uranium-238. The average radium concentration of the sandy soil of the Laak basin is 13 Bq/kg (4), somewhat lower than the world average of the earth's crust of 40 Bq/kg (1). Radium-226 decays with a half-life of 1600 years in radon-222, which is a noble gas and thus chemically inert. It can move freely in the earth's crust and in building materials and may eventually reach the atmosphere. Radon is in its turn radioactive, with a half-life of 3.82 days, so that its short-lived decay products are present in the air delivering a dose to the lungs.

A radium contamination of the environment can expose the population to ionizing radiation in several ways. The dominating modes of exposure are:

- *External irradiation* close to the contaminated area.
  The gamma dose rate at one metre above ground with a homogeneous radium concentration of 1000 Bq/kg, in equilibrium with its decay products, is 0.49 µSv/h (4).
- *Internal exposure* through the consumption of contaminated food.

The ingestion dose conversion coefficient of ^{226}Ra for adult members of the public is 2.8 10^{-7} Sv/Bq (5).
Inhalation of the short-lived radon decay products.
The decay products of radon are chemically active and remain long enough in the lungs for decay to ^{210}Pb. In this process, two α-particles are emitted, originating from ^{218}Po and ^{214}Po. The dose conversion coefficient of ICRP 65 for indoor exposure is 2.4 nSv/h/(Bq/m³) (6).

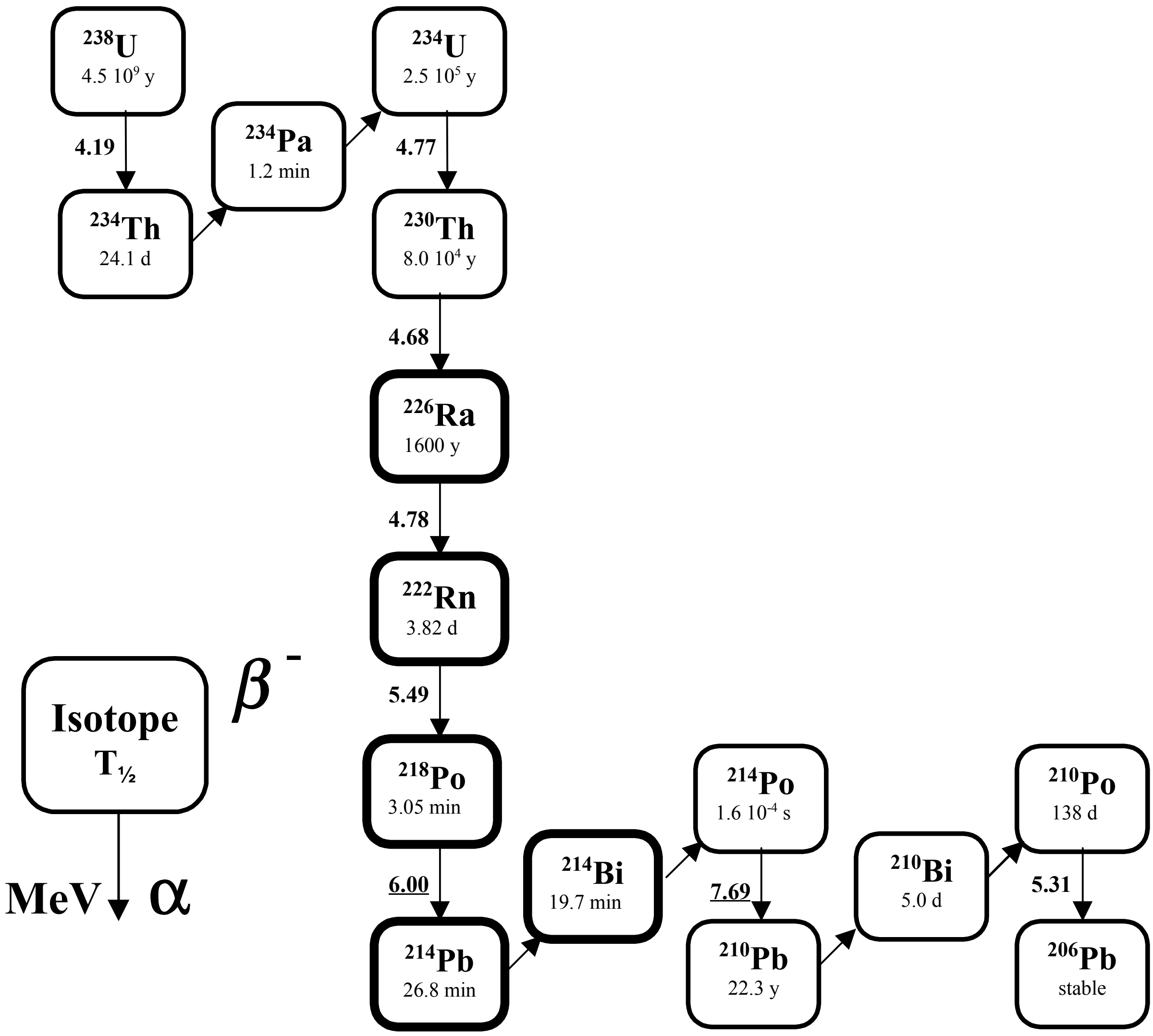

Figure 3. The (natural) radioactive uranium series. The inert radionuclide radon-222 is formed by alpha decay from radium-226. The cascade of radioactive transformations ends with the stable nuclide, lead-206.

METHODS

The radium contamination of the river banks was mapped over a distance of 20 km from the discharge points to the merging with the river the Grote Nete. Gamma dose rates were measured every 50 m perpendicular to the Laak with portable detectors. Measurements were done at both banks at a distance of less than 1 m of the water, and subsequently every 2 m until the background level was found or until obstacles prevented access. The flooding zones were measured in loops, moving away from the contaminated river banks until background levels were reached, taking as much values as needed to effectively map the contamination of the zone. The measurements were performed close to the vegetation at about 10 to 20 cm above the ground. They were done with portable gamma monitors of the firm Automess, type 6150 AD6 fitted with the plastic scintillator detector 6150 AD-b. These monitors have a measuring range between 50 and 99000 nSv/h.

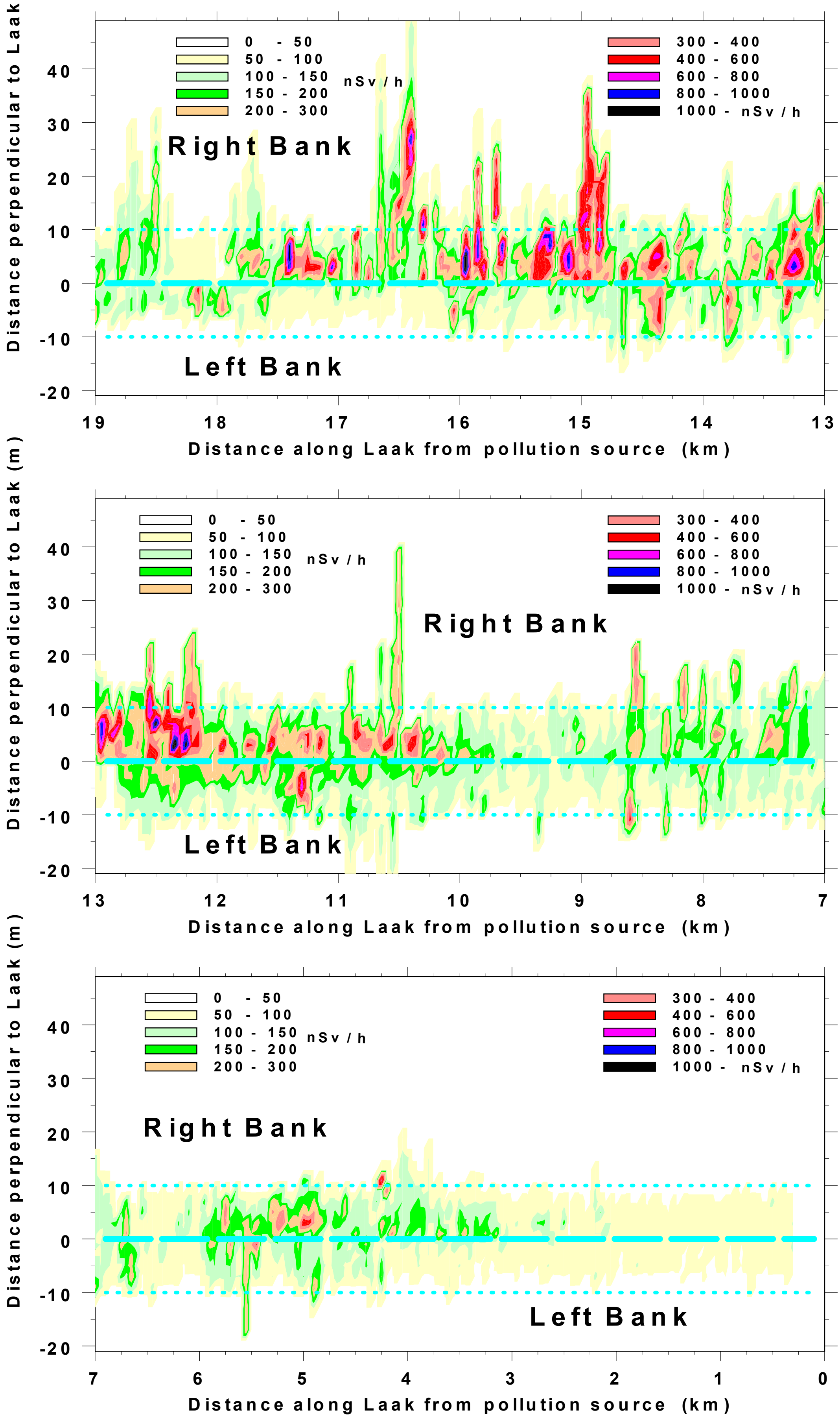

Figure 4. Representation of the radium contamination of the river banks. Gamma dose rates were measured every 50 m perpendicular to the Laak until background levels were reached.

Soil samples were taken regularly, mostly at a local maximum. Their ^{226}Ra concentration was determined at the laboratory by means of high-resolution gamma spectroscopy. The samples were taken under the vegetation close to the surface to make the link with the local dose rate measurements. Due to the rough nature of the terrain and the sometimes dense vegetation the measurement campaign took two weeks with teams of three persons on each side of the river.

RESULTS

The results of the dose rate measurements of the river banks are shown in figure 4. The Laak is represented by a straight line in the middle of the figure and flows into the Grote Nete at kilometer 19. For clarity, distances perpendicular to the river are magnified. Enhanced dose rates are found everywhere along the Laak between the discharge points and the merging with the Grote Nete. The contamination is mostly confined to a narrow zone of about 10 m on both sides of the river, caused by the periodic dredging with mud deposition on the banks. The right bank is more contaminated than the left bank because there are more pastures on the right bank making it easier to deposit the sludge on this side of the river. The measured dose rates vary between the low natural background of the region, 50 to 80 nSv/h, and 1000 to 2000 nSv/h. The dose rates differ often over an order of magnitude within a few metres. The most contaminated zones are found 12 to 17 km downstream from the plant discharges, where the river follows a very meandering path.

The soil samples that were taken mostly at local maxima had typically radium concentrations of a few thousands of Bq/kg. The average value of the samples taken at the left bank is 2800 Bq/kg and at the right bank 3800 Bq/kg with a standard deviation of 1800 Bq/kg for both values. This is two orders of magnitude more than the average radium concentration of the soil of this region, 13 Bq/kg (4).

EVALUATION OF THE CURRENT POPULATION EXPOSURE

External irradiation close to the contaminated area

The surface area of the contaminated banks was calculated from the gamma measurements assigning a surface of 100 m² (2 m x 50 m) to each point. The combined results for the left and right bank are shown in figure 5. Approximately 22 ha are contaminated above background (more than 100 nSv/h) and 12 ha have a dose rate above 150 nSv/h.

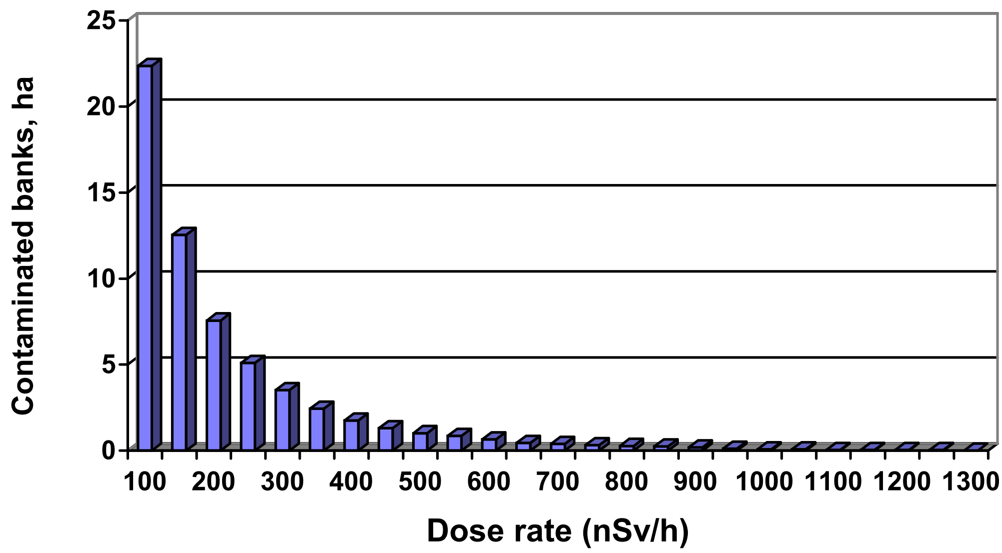

Figure 5. Surface area of the banks contaminated above the indicated dose rate.

The contaminated river banks are freely accessible to the public. The current external exposure of the critical groups is estimated at a few hundreds of µSv/year depending largely on the proposed residence times and exposure levels.

Internal exposure through the consumption of contaminated food

Pastures and maize for animal feeding are currently the only cultures along the contaminated banks of the Laak. This additional step in the food chain reduces the internal doses to a few tens of µSv/year. The transformation to arable farming with crops for direct human consumption would increase the exposure of the critical group through the food chain to a few hundreds of µSv/year.

Inhalation of the short-lived radon decay products

The exhaled radon is very diluted in open air. The contamination of the Laak has the shape of a line geometry limiting the increase of the radon concentration to a few Bq/m³. Transformation of the contaminated banks to a residential area would undo this favorable situation. The radon concentration in the dwellings could reach very high values resulting in doses to the residents that could exceed 20 mSv/year, the limit for radiation workers.

CONCLUSION

Enhanced dose rates up to 2000 nSv/h were found along the banks of the Laak downstream from the discharge points of a large phosphate plant. The periodic dredging and deposition of the sediment on the banks caused the contamination. The current population exposure was assessed and doses of less than 1 mSv/year were estimated to members of the critical groups from external irradiation, the consumption of contaminated food and the inhalation of the short-lived radon decay products.

Although the current agricultural activities, with only crops for dairy farming, pose no problems, the transformation of the contaminated banks to a residential area could lead to indoor radon exposures exceeding 20 mSv/y. Also the transformation from dairy farming to arable farming with crops for direct human consumption would increase the exposure of the critical group through the food chain to a few hundreds of µSv/year. Measures in the sphere of town and country planning have to be taken, preventing a change in the use of these grounds. Both local, regional and federal authorities have an important role to fulfil in this.

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Annales de l'Association Belge de Radioprotection, Vol 25, n°3 pp 203-204 P-3a-139

In Vivo Measurement Of Internal Contamination In The Low Energy Range: A Study Of The Background Effect

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The assessment of internal contamination by the direct methods requires special attention when the in vivo detection of low energy photon emitters (10 - 100 keV) is concerned. The usually used shielding rooms, by the production of fluorescence, are often increasing the continuum of the background instead of reducing it, leading to an increase of the detection limits.

Experimental researches comparing different types of detectors in different environmental conditions have shown that the measurements in the low energy range can be performed out of a shielding room when the background is controlled. The results of the study show that the background can be reduced without the use of a heavy shielding construction in the quantitative assessment of body burden. The analysis shows that the best results always are obtained when the detector's volume is adapted to the investigated energy. In this case, an array of multiple small diodes will allow a better assessment of the deposition in the body, the localisation of the deposition and a better quantification after the correction of the calibration factors.

The use of new types of small detectors (semiconductor type: silicon, CdZnTe...) also allows a good positioning of the detectors on the measured organ leading to a better efficiency and a better accuracy.

When the measurement can be performed out of a special room, the counting duration can be extended to several hours. Extrapolations from the experiments have shown that arrays of diodes can produce results with the same detection limits as the HPGe or phoswich detectors presently used with the advantageous of a possible control of the deposition pattern.

Using this technique can develop different applications in Radiation Protection and in nuclear medicine. They include the metabolism studies of radionuclides, the detection of local deposition in case of inhalation, ingestion or injection and the control of wound immediately after the accident.

Annales de l'Association Belge de Radioprotection, Vol 25, n°2, p 205 P-3b-149

The BDT Bubble Neutron Detector For Personal Dosimetry

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Twenty years after their discovery, the bubble detectors are progressing from a laboratory tool towards a workfield dosemeter used for neutron dosimetry. The high sensitivity, direct readability and near dose equivalent response make them at present the best alternative for personal neutron dosimetry. Most popular is the BD-PND bubble detector, available from Bubble Technology Industries. This detector has a threshold around 100 keV, so in neutron fields with an important thermal contribution, it is necessary to supplement the dose measurements with the BDT bubble detector. This one has a different chemical compostion so that it measures mainly thermal neutrons.

Following a thorough study of the BD-PND detector we have investigated the characteristics of this BDT detector. We determined the energy dependence using mono-energetic neutrons and radionuclide sources and checked the influence of a phantom. Using thermal and ^{252}Cf neutrons we also determined the angular dependence of the BDT. All these results learned us to what extent the BDT gives the personal dose equivalent $H_p(10)$. Another important point that we checked was the evolution of the sensitivity with time. Just as the BD-PND detectors, the response of the BDT detector is influenced by the temperature. Also here the temperature compensating system is the limiting factor in the use of these bubble detectors. We determined the change in temperature compensation with their age, again using both thermal and ^{252}Cf neutrons.

All these tests lead to the conclusion that the BDT bubble detector is a good complement to the BD-PND detector for personal neutron dosimetry, but that care has to be taken when they are used in strongly thermalised neutron fields at high temperatures and large angles.

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Radiological and Economic Impact of Decommissioning Charged Particle Accelerators

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INTRODUCTION

Currently there are about 250 particle accelerators (not including medical linear accelerators) in operation in the European Union (EU). These are used for radionuclide production, radiotherapy and research.

For biological shielding these accelerators are housed in buildings with thick concrete walls. During the operational life of the accelerator those walls become radioactive over time, just like the accelerator itself and all infrastructures in the irradiation rooms. When considering decommissioning of these accelerators, large amounts of low level solid radioactive waste have to be taken into account. However, practically no data on the type of activation, the level of activation or the depth of activation of the concrete and metallic infrastructure existed. For this reason DG ENV/C3 of the European Commission (EC) launched a research project in 1997 focusing on "The evaluation of the radiological and economic consequences of decommissioning particle accelerators". The contract (B4-3070/97/00024/MAR/C3) was granted to the Cyclotron Department of the Vrije Universiteit Brussel (VUB). The objectives of the project can be summarised in 5 main points: inventory of the decommissioning problem of accelerators in the EU, characterisation of the activation of 3 reference accelerators and their shielding, estimation of dismantling techniques, costs and potential waste volumes evaluation and recommendations for prevention.

INVENTORY OF ACCELERATORS

To estimate the economic and radiological consequences of decommissioning the EU particle accelerator park (not medical), it is necessary to have a correct idea of the total number of accelerators present in the EU, their operational status, their main uses and the corresponding beam loads and beam energies. In addition to these basic data, information on the shielding and the metal infrastructure of the installation allows a more reliable view of the radiological situation of the accelerator and its environment. As the information needed is rather detailed, the institutes themselves were contacted. Based on an extensive survey of the available literature and with the help of the local authorities, a preliminary address list was composed, containing 226 installations. A questionnaire was sent to all the identified installations, containing questions dealing with the 5 topics mentioned in Table I. This questionnaire was also published on the Internet allowing online access and immediate electronic processing of the information supplied. After sending several reminders, a total of 91 accelerator facilities supplied (part of) the information needed. This information was collected in a database from which a number of important conclusions could be drawn.

Table II compares the number of facilities responding to the questionnaire with the total number of facilities located in each country. From this table we conclude that Sweden, Italy and Holland, who have a rather large number of accelerators, have done a serious effort in replying to the questionnaire. On the other hand, Germany and especially France have failed largely in replying to the questionnaire, although together they account for over 50% of the EU-accelerators.

The distribution of the answers to the question on the type of shielding used was the following: no answer 26, fully movable shielding 5, massive shielding walls 28, massive and movable shielding 32. For most of the accelerators using a mixture of massive and movable shielding, the ratio of massive to movable is larger than 1. As a result, it can clearly be stated that massive shielding is used in most cases what can result in additional

Table I. Layout of the questionnaire

	Topic	Description
1	General Information	Deals with general data on the institute, the operator, the reporter and the accelerator.
2	Accelerator Data	Asks for more details on the accelerator characteristics.
3	Irradiation and Activation Data	Probes more in detail to the risks of activation of the infrastructure or shielding using information on the beam history, history of the shielding and composition of the shielding and infrastructure.
4	End of Life of Accelerator and Decommissioning	Checks to what extent the institutes are aware of the consequences of decommissioning a particle accelerator and of the national regulations.
5	Future Plans	Contains information on possible new accelerators.

Table II. Comparison between number of answers received and number of questionnaires sent per country

Country	N° located	N° received	Ratio [%]
Greece	1	1	100
Portugal	1	1	100
Finland	6	5	83
Sweden	11	8	73
Holland	18	12	67
Italy	18	9	50
Denmark	4	2	50
Belgium	15	7	47
United Kingdom	20	7	35
Germany	91	30	33
Spain	4	1	25
France	35	8	23
Ireland	2	0	0
	226	91	40

problems during the decommissioning phase. The database also indicates that Ba, Fe or other heavy elements are seldom used in the walls of accelerators. The effect of long-lived radionuclides produced on these elements is hence minimal.

With respect to activation of the concrete shielding, the metal infrastructure and the machine parts, the information in Table III was supplied. A broad range of values can be noticed, clearly indicating that considerable differences in activation can be expected in the different installations.

Table III. Reported specific activation in concrete and metal parts

concrete (for 3 cyclotrons)	^{152}Eu	700	-	12000	Bq/kg
	^{60}Co	80	-	8000	Bq/kg
	^{46}Sc	800	-	900	Bq/kg
metal infrastructure (for 3 accelerators)	^{60}Co	0.10	-	100	kBq/kg
	^{54}Mn	0.09	-	380	kBq/kg
	^{65}Zn	1.64	-	170	kBq/kg
machine parts	^{60}Co	32	-	5000	kBq/kg
	^{54}Mn	0.9	-	1000	kBq/kg
	^{22}Na	1000	-	10000	kBq/kg
	^{57}Co	0.25	-	100	kBq/kg

The low number of replies received to these questions furthermore indicates clearly the lack of awareness of a long-term waste problem around accelerators. This demonstrates that the problem of decommissioning particle accelerators is strongly underestimated.

Another conclusion from the database is that we are dealing with an ageing accelerator park with a majority of installations in the public or health sector. From several contradictory answers can be concluded that the regulatory framework is generally unknown by users of accelerators. In addition to the underestimation of the problem, the existence of a decommissioning plan is exceptional and almost no institutes foresee provisions.

SELECTION OF 3 REPRESENTATIVE ACCELERATORS

The large number of accelerators in the EU and their variety makes that the activation can only be analysed in detail through a limited number of representative case studies. When excluding the medical linear accelerators used for radiotherapy, the database learns that the accelerators can be subdivided into 4 classes (see Table IV).

Class 1 accelerators do not pose an activation risk as the energy delivered is below the threshold for the large majority of nuclear reactions. Hence the possibly generated secondary particle fluxes are inexistent or very low, producing negligible activation of the shielding. No systematic investigations have been carried out for accelerators in this class.

Class 2 contains the medium energy cyclotrons and linear accelerators, used for research, radionuclide production and radiotherapy. This group represents over 50% of the accelerator park of the EU. As the energy delivered is high enough to produce activation by secondary particles and as a high particle flux is characteristic

Table IV. Classes of accelerators.

Class	Description
0	Radiotherapy linear accelerators (estimated 1200 machines in the EU, not addressed in the questionnaire, direct contact with manufacturers established, practical decommissioning experience available).
1	Low energy (2-10 MeV) linear and electrostatic accelerators, grouping essentially a score of Van de Graaff, tandem and similar accelerators.
2	Medium energy (10-100 MeV) proton, H^- or multiple particle (including heavy ions) cyclotrons and linear accelerators mainly used for physics research (also injectors) and radionuclide production. As confirmed by the results of the questionnaire this class represents over 50% of the accelerator park in the EU.
3	High energy (100-300 MeV) proton cyclotrons or synchrocyclotrons and high energy linear accelerators (used for basic physics research, frequently as spallation neutron sources generating hence high neutron fluxes and activation). About 6% of the accelerator park.
4	Very high energy synchrotrons and storage rings up to several GeV, used in high energy physics, making up 12% of the accelerator park of the EU.

for these machines, problems with activation of concrete shielding and metal infrastructure can be expected. As a representative of this class, the cyclotron of the VUB was chosen. It is a variable energy multi-particle cyclotron with 43 MeV maximal proton energy for 100 µA maximal beam intensity. The utilisation of the different irradiation rooms includes semi-commercial radionuclide production and research, yielding a wide range of activation.

In class 3 we find the high energy cyclotrons and linear accelerators used for basic research, frequently as charged particle based spallation neutron source. The 200 MeV electron linear accelerator of the Institute for Reference Materials and Measurements (IRMM/JRC) in Geel (Be) was chosen as representative in this group. The very high neutron production rate around this accelerator will probably yield the upper limit of activation to be expected.

The group of very high energy synchrotrons and storage rings forms class 4. The complexity of these installations and the difficulty to have access to the facilities for an extensive investigation campaign limited the possibility of a real choice for a representative case in this class. Due to the coincidence of our investigation with the start of a local decommissioning study, the large 6 GeV proton synchrotron SATURNE of the CNRS-CEA (Fr) was chosen as representative in this class.

EXPERIMENTAL DETERMINATION OF THE ACTIVATION STATUS

Activation of the Concrete Shielding:

Activation of the concrete shielding is one of the most important problems at decommissioning. Although the principles of activation are well understood, only limited studies on the relation between accelerator characteristics and activation levels are published (1, 2, 3, 4, 5). It was therefore necessary to obtain detailed quantitative information on the radioactivity induced in the concrete shielding. The activation of the concrete shielding is produced by secondary neutrons and photons. These are generated in collisions of the primary particles with the accelerator itself, during the beam transport and through interaction with the target.

At the 3 facilities core samples of the concrete shielding were taken with a boring device using a diamond-drilling tool of 50 mm diameter. The closed circuit of the cooling system of the boring device enabled an almost 100% recuperation of liquid effluents in a tank. The risk of dust contamination or dust intake was low due to the wet drilling process preventing the airborne spread of activity. After a few hours the water in the tank could be separated from the sediment, which has to be disposed of as low activity nuclear waste. All the removed concrete cores were then cut in 5 cm long samples (± 0.2 kg) by a diamond saw. These concrete samples were analysed with a high-resolution γ-spectrometer (HPGe-detector) using correction for self-absorption and measurement geometry. When present, pieces of the reinforcement steel were removed and measured separately.

For the VUB cyclotron a total of 96 drillings were performed. The drilling campaign was concentrated in the accelerator vault and in irradiation room 2, as these rooms have been the most intensely irradiated (78% of the integrated beam load was put on target in irradiation room 2). At the IRMM linear accelerator a total of 61 drill cores were taken, mainly concentrated in the target hall. Due to the very localised beam losses of a synchrotron, only 2 well-defined regions (around the beam extractors) of the shielding of the SATURNE accelerator can be activated. Hence 50 drillings were performed, distributed over these 2 regions.

Table V shows the radionuclides detected in the concrete after γ-spectrometric analysis. The activation of trace elements of metals (e.g. europium) in sand yield medium lived radionuclides. These are mainly created by neutron capture with high cross sections and by threshold reactions with lower yield. Because of the high ^{151}Eu

Table V. Radionuclides identified in the concrete shielding.

Radionuclide	Possible Reaction	Cross section	Half life
^{152}Eu	^{151}Eu (n,γ) ^{152}Eu	9198 barn	13.33 years
^{154}Eu	^{153}Eu (n,γ) ^{154}Eu	312 barn	8.8 years
^{134}Cs	^{133}Cs (n,γ) ^{134}Cs ^{134}Ba (n,p) ^{134}Cs	29 barn 9 mbarn at E_n = 16 MeV	2.06 years
^{60}Co	^{59}Co (n,γ) ^{60}Co	37 barn	5.3 years
^{46}Sc	^{45}Sc (n,γ) ^{46}Sc	27 barn	83 days
^{133}Ba[a]	^{132}Ba (n,γ) ^{133}Ba	7 barn	10.5 years
^{54}Mn	^{55}Mn (n,2n) ^{54}Mn ^{54}Fe (n,p) ^{54}Mn	910 mbarn at E_n = 18 MeV 590 mbarn at E_n = 10 MeV	312 days
^{22}Na	^{23}Na (n,2n) ^{22}Na ^{27}Al (n,2p4n) ^{22}Na	40 mbarn at E_n = 15 MeV 10 mbarn at E_n = 25 MeV	2.6 years
^{137}Cs	^{136}Ba (n,γ) ^{137m}Ba → ^{137}Cs ^{137}Ba (n,p) ^{137}Cs	0.4 barn 3.7 mbarn at E_n = 16 MeV	30 years

[a] only detected where “barite” concrete is used

cross section for thermal neutron capture and the long half life of ^{152}Eu, this nuclide is present in large quantities in the activated shielding. At the VUB cyclotron and the IRMM linear accelerator, the highest detected specific activity for ^{152}Eu was 11 kBq/kg, respectively 90 kBq/kg. At SATURNE the 2 most important radionuclides present are ^{133}Ba and ^{22}Na due to the use of barite concrete and concrete with high Na content. A maximal specific activity of 1.8 kBq/kg was measured for ^{133}Ba. These values are considerably higher than the different proposed clearance levels, which are all between 100 Bq/kg and 1 kBq/kg as discussed further in this text.

For the different radionuclides found in the concrete, in-depth activation profiles were drawn. Figure 1 shows such a typical profile from irradiation room 2 of the VUB cyclotron, from which can be concluded that after an activity build-up in the first 15 cm, a quasi-exponential decay of the activity occurs. This behaviour is found for nearly all profiles of both VUB cyclotron and IRMM linear accelerator. The in-depth activation profiles of the SATURNE synchrotron does not show the build-up. In addition to the γ-analysis of the concrete, the tritium content of several samples was determined. Small chunks of concrete originating from the 3 reference accelerators were ground and sieved for this purpose. Samples of 0.1 g were prepared and heated to 1100°C. The water liberated from these samples was mixed with a scintillation liquid. Specific activities for ^{3}H up to 380 kBq/kg were found and correlate well with the corresponding ^{152}Eu activation. Tritium is probably produced by spallation reactions that frequently occur around high energy accelerators. However further studies are required to obtain a satisfactory explanation for this ^{3}H-formation in concrete.

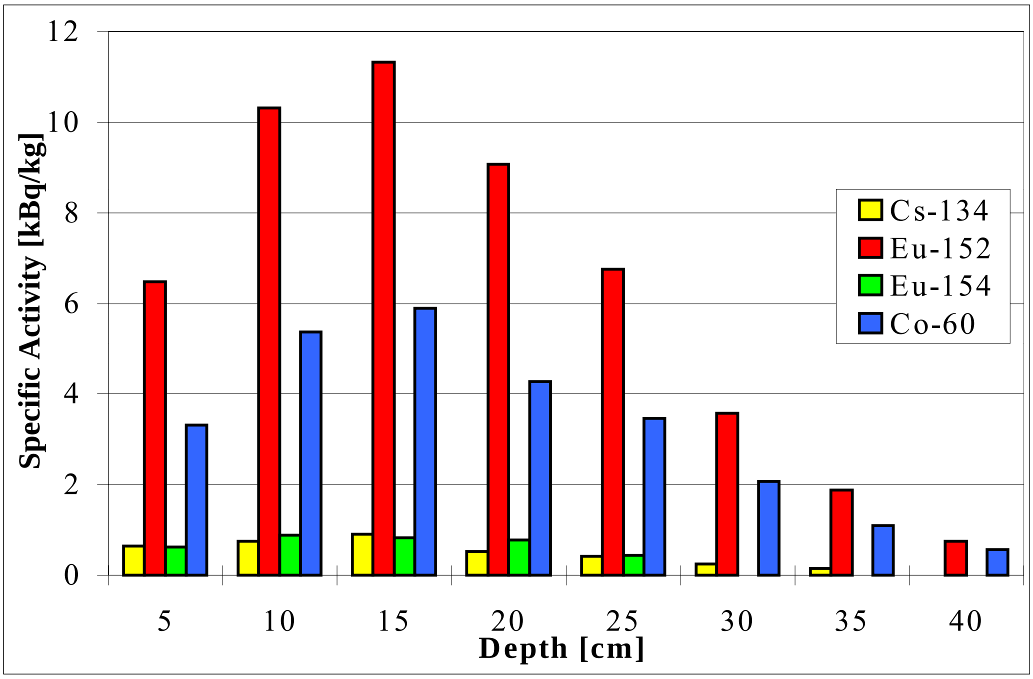

Figure 1. Typical in-depth activation profile (irradiation room 2 - VUB cyclotron).

Activation of Metal Parts:

At all three facilities, a sampling campaign of different kinds of metal parts (aluminium, stainless steel, plain steel, copper, brass and galvanised steel), originating from the accelerator rooms, target rooms and experimental halls, was carried out to investigate the activation. The samples were taken from both the machine itself and the surrounding infrastructure which can be activated by neutrons.

The measurements of the infrastructure parts of the VUB cyclotron show considerable amounts of ^{60}Co, ^{65}Zn and ^{54}Mn with specific activities between 1 and 200 kBq/kg. Most aluminium parts have a specific activity below 1 kBq/kg, while a few parts of steel and stainless steel constructions have a specific activity above 200 kBq/kg.

Attention is drawn to the fact that the yokes of the main magnets and switching magnets, which are made of low Co steel (Co is only present as a possible impurity), are only slightly activated. Highly activated machine parts are the deflector and its support structure and the accelerating Dee structures. Both structures are activated far above 1 MBq/kg and have to be removed as soon as possible after shutdown, nevertheless allowing for decay of short-lived nuclides.

The reinforcement rods of the concrete walls were activated up to 300 kBq/kg. Analogous results were found from the measurements at IRMM and CEA.

Interpolation and 3D distributions:

The final goal of this work is to estimate the total decommissioning cost of particle accelerators. For this purpose information on the activation products is not only needed in the sampling points, but as a 3D distribution over the entire walls of the building and in the depth of these walls. A technique was developed for the extrapolation from a few simple points to an activation distribution over an entire room. This extrapolation technique is based on the conclusion that the main parameters influencing the specific activity $A^i(x,y,z)$ produced by secondary neutrons in a certain point of layer i of a concrete wall are:

- the source-concrete distance for the point of interest $R^i(x,y,z)$;
- the relative number of neutrons able to produce activation $\varphi^i(x,y,z)$ in the direction of interest.

The activity in a layer of concrete $A^i(x,y,z)$ can hence be written as:

$$A^i(x,y,z) = A_0^i \varphi^i(x,y,z) \frac{\left(R_0^i\right)^2}{\left(R^i(x,y,z)\right)^2} \tag{1}$$

where A_0^i is a know activation level in layer i of the concrete (based on the measurement of the core samples) and R_0^i is the corresponding source-concrete distance. Based on the measurements of the other core samples, discrete values for the function $\varphi^i(x, y, z)$ can be calculated:

$$\varphi_j^i = \frac{A_j^i (R_j^i)^2}{A_0^i (R_0^i)^2}$$

Assuming a cylindrical symmetry and using a 3rd order polynomial fit, the unknown function $\varphi^i = \varphi^i(\theta)$ can be found. Based on this result and on (1), the specific activity in any point (x,y,z) of concrete layer i can be calculated. Repeating this for all layers, the activation level in any point of a concrete shielding can be found for every radionuclide of interest.

A typical result of this interpolation is shown in Figure 2. The activation level determines the colour scale code for each voxel at that point. It changes between 0 and 10 kBq/kg. The voxel size was chosen at 10x10x10 cm^3. For each wall the highest activation levels are found at the points opposing the neutron source (target). From the figure it is also clearly visible that, instead of the first layer of concrete, the second one contains the highest specific activity, which is fully in accordance with the activation profiles found. Further information on this interpolation scheme can be found in (6). These interpolations yield the input to the cost evaluation procedures.

Medical Linear Accelerators:

Although the activation of and around linear accelerators used for radiotherapy is generally considered very low, the total amount of radioactive waste can be considerable due to the important number of accelerators concerned (>1200 in EU, 300 new accelerators or replacements installed each year in Europe).

Concrete samples were taken from the upper 5 cm of the floor under the treatment table of a medical linear accelerator (20 MeV electrons, 18 MV photons, 15 years of operation), showing no significant activation above background level. It can hence be concluded that the neutron fluxes produced during the operation of this type of accelerator are too low to activate this upper layer of the concrete shielding. Our measurements around the other types of accelerators show that the maximum of activation of the concrete is found at deeper layers. However these measurements indicate that most likely no concrete activation of importance will be found around

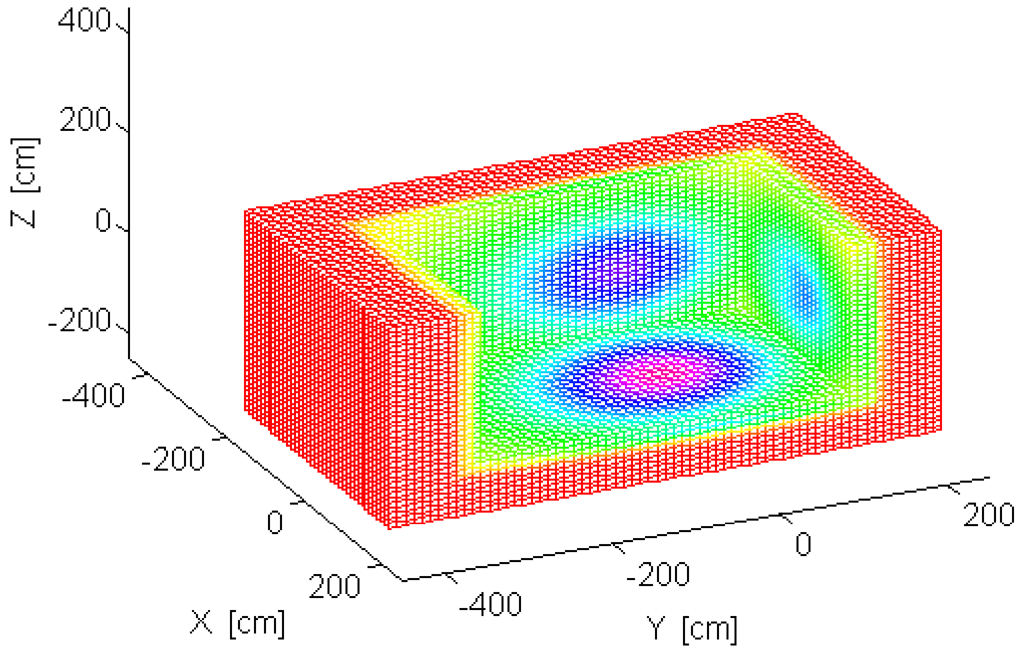

Figure 2. 3D distribution of the specific activity of ^{152}Eu in irradiation room 2.

medical linear accelerators. Further details can be found in (7).

To evaluate the activation of machine parts samples were taken from a 23 MeV accelerator. Measurement revealed activation levels up to 26 MBq/kg of ^{181}W and 70 kBq/kg of ^{60}Co (both in the conversion target). The relative short half life of ^{181}W would allow release of the material after 5-6 years, but due to the presence of ^{60}Co, more than 40 years of cooling time is necessary to bring its value under 300 Bq/kg. Presence of Sb-nuclides in activated lead pieces could drastically increase these cooling times.

Important amounts of ^{110m}Ag were found in the soldering material used in the wave-guides, requiring cooling times of over 100 years. The other materials pose less long term waste problems due to the short half lives. During the dismantling of the 20 MeV linear accelerator mentioned earlier, a total of 400 kg metal parts was disposed of as (low level) nuclear waste.

DISMANTLING TECHNIQUES

Some remarks on possible techniques for dismantling particle accelerators are presented here. The objectives of these techniques include minimisation of the exposure of personnel and minimisation of the generated amounts of waste. Dismantling techniques should be selected and used so that activated plant and equipment components are removed in a controlled way, avoiding mixing with non-activated wastes where possible. If possible unconditional and conditional recycling are to be preferred to the apparently more simple method of disposal as radioactive waste, not least of all due to economic considerations.

The selection of cutting techniques used for dismantling has to be optimised from the point of view of radiation protection, secondary waste generation and cost-efficiency. The radiological protection requirements to avoid any unnecessary radiation exposure necessitate cutting techniques that allow high cutting speeds, automatic operation or remote control. Simple, proven techniques can also meet these requirements. Therefore, during decommissioning, conventional cutting techniques will be mainly used. These have to be adapted however to the special conditions, such as the confined space and possibilities of transporting the cut-away components.

Moreover, the cutting techniques have to be adapted for the activity levels and materials of the components that are to be cut up. Special attention must be paid in this respect to the avoidance of dust (aerosols) and to their safe containment in cases where they are unavoidable.

A very wide range of techniques is available to allow a clean dismantling of concrete structures, i.e. with the least possible generation of dust: sawing, wire-cutting, circular sawing, chain sawing, drilling, core drilling, cutting with special hydraulic pincers, operated manually on gripper arms and thermal exposure of reinforcing steels by means of electric resistance heating (8).

Where dust and gas formation can be well contained it may be advantageous to use controlled, gentle detonation techniques. This is the case in the vaults of particle accelerators.

COST EVALUATION

With respect to cost evaluations, no conformity exists between the different member states of the EU. There are large differences in labour costs and waste management costs. In addition to this, currently no international consensus on clearance levels exists as both can be seen from Table VI. In Germany, for instance, a strict clearance regulation is in place, using 100 Bq/kg as general clearance level, whereas Great Britain is applying 400 Bq/kg. France has no clearance levels but has a system based on "zoning", where waste from "non contaminating" zones can be treated as conventional waste, while waste from "contaminating" zones can be eliminated as very low level waste as long as the specific activity is lower than 100 kBq/kg. In addition to this, Germany allows melting of metallic waste in the conventional industry as long as the specific activity remains below 1 kBq/kg. Furthermore there is the basic safety standard of the EC where nuclide specific levels are proposed for exemption, which can be considered an upper limit for all clearance levels in case total quantities are limited. The EURATOM Art. 31 Expert Group has recently provided further guidance.

Table VI. Clearance criteria and labour costs in the EU

	Germany	Great Britain	France
Clearance (γ/β-activity)	0.1 kBq/kg	0.4 kBq/kg *	no clearance; managed as VLLW
Clearance for melting in the conventional industry	1 kBq/kg	-	
Labour cost in industry **	42.25 €/h	22.97 €/h	33.80 €/h
Total Cost in industry ***	55.72 €/h	27.34 €/h	49.28 €/h

* Limits given by the Radioactive Substances Exemption order issued in 1996.
** Values for 1995; the values in ECU have been converted into € by a factor 1.
*** Direct + indirect costs, but without social security paid by the employer.

With all these different situations, a choice of 7 different decommissioning scenarios was made: an immediate dismantling in the French, German and British situations (also using local labour and waste treatment costs), an immediate dismantling with these 3 cost assumptions using the clearance levels recommended by the EC (only for VUB cyclotron) and a deferred dismantling scenario in the British case. In this last case cooling times for the concrete of 70 years (VUB cyclotron), 105 years (IRMM linac) and 75 years (SATURNE) are needed to reach the clearance level applied in UK.

The resulting waste volumes (in tonnes) are presented in Table VII and the total decommissioning costs in k euro are found in Table VIII. It can be concluded that:

- the British scenario seems to be the most cost-effective one, not only due to the higher clearance levels, but also mostly due to lower labour and waste management costs;

- the French scenario provides much lower costs compared to the German one owing to the fact that the costs for very low-level waste management are assumed to be very low (1/10 of the costs for low-level waste in France and about 1/25 of those costs in Germany). Due to the high amounts of very low level waste in the French scenario, the results of this evaluation could thus be quite different in case of cost increasing for this type of waste in France;
- costs for deferred dismantling are lower than those for immediate dismantling. Nevertheless, this scenario needs to be handled carefully because the costs for long-term safe store depend greatly on the context within which the installation needs to be maintained. The feasibility of maintaining the installation during such long periods under regulatory control within a changing surrounding needs also to be evaluated carefully. Furthermore, the implementation of the EC-recommended exemption system, as an assumption for maximal clearance levels, competes with the deferred scenario from the point of view of decommissioning costs;
- the use of these maximum clearance levels would considerably reduce the costs for decommissioning accelerators due to the very low level of activity in the structures. This illustrates the sensitivity to the adopted clearance levels. This conclusion is specific for accelerators and cannot be extended to other types of facilities where the ratio of very low level waste to other radioactive waste could be quite different.

PREVENTION OF ACTIVATION

The results of the three extensive case studies presented here show that the major part of long lived radionuclides present in the waste around particle accelerators are due to neutron induced reactions. The resulting radiological burden to workers during exploitation of the accelerator and the immense economical consequences of dealing with the nuclear waste at decommissioning asks for important measures to reduce or prevent activation of machine parts, infrastructure or shielding in a cost effective way. The improvement in design of accelerators and changes in the concept of set-ups for lowering the neutron production particularly yield a reduction of activation and of the waste to be disposed off at dismantling. Equally important are measures to minimise the interaction of these neutrons with materials prone to activation. A judicious choice of construction techniques adapted to the specific requirements for zone-wise dismantling of nuclear installations is necessary. Use of local shielding around target stations and modular construction of main shielding walls are recommended options.

In the field of particle accelerators, radiation losses and secondary particle generations are directly linked to beam losses. Of course, the minimisation of beam losses has always been an important design criterion for accelerators. This is not just in view of its decommissioning, but mainly in view of optimising the beam current and of the very important aspect of maintenance. The success of using H^- acceleration in a deep valley magnetic structure (large variation between highest and lowest magnetic field along one particle orbit) is exactly based on these factors. Another remark concerns the future of particle accelerators, and in particular of proton accelerators. It becomes more and more evident that future applications of particle accelerators will require strongly increasing amounts of beam power. In such a context it is clear that a very strong reduction of all beam losses is a vital element in the design of new particle accelerators. The passage through a particle accelerator has 3 phases: injection, acceleration and extraction. All 3 phases may be responsible for beam losses, and each needs to be optimised.

Unavoidable radiation losses and generation of spurious neutron fluxes always exist at the collimators, target stations or beam dumps for any type of accelerator. Prevention or diminishing the amount of activated material in the decommissioning waste is however possible by a choice of materials and by adapting the shielding concept.

For a number of applications the use of adapted materials for collimators, beam stops or parts of target holders and irradiation set-ups can greatly reduce the unwanted secondary neutron production. The best example is the replacement of Cu and stainless steel by pure carbon at low energy. The threshold at 18 MeV of the (p,n) reaction guarantees a lower total neutron yield for incident beams up to about 25 MeV proton energy and nearly no long lived activation products. Even at higher energies, the yield of secondary neutrons is always lower for C than for other materials used for the same purpose. A possible negative aspect is the generation of high activities of short lived β^+ emitters limiting rapid access to the vicinity of the irradiated C-parts and the inferior properties concerning degassing and ultimate vacuum. Similar considerations exist for the use of aluminium.

Table VII. Waste volumes for the different scenarios (in tonnes)

	SCENARIO	WASTE	ACCELERATOR		
			VUB	IRMM	SATURNE
Immediate	German scenario	LLW concrete	686	1468	612
		LLW metals*	5	27	1.3
	French scenario	LLW	14	28	1.3
		VLLW	2105	3130	1180
	British scenario	LLW	648	850	342
Deferred	British scenario	LLW	0.005	6	1.3
Immediate	German costs, EC-recommended clearance levels	LLW concrete	74	X	X
		LLW metals*	5	X	X
	French costs, EC-recommended clearance levels	LLW	13	X	X
		VLLW	78	X	X
	British costs, EC-recommended clearance levels	LLW	91	X	X

* In the German scenario, radioactive metal is assumed to be molten and recycled in the Siempelkamp facility, which reduces the amount of LLW.

Table VIII. Decommissioning costs for each of the scenarios (in k€)

	SCENARIO	ACCELERATOR		
		VUB	IRMM	SATURNE
Immediate	German scenario	7700	16010	6870
	French scenario	4340	6320	1445
	British scenario	3550	4540	1820
Deferred	British scenario	980	1340	875
Immediate	German costs, EC-recommended clearance levels	1545	X	X
	French costs, EC-recommended clearance levels	810	X	X
	British costs, EC-recommended clearance levels	830	X	X

As already mentioned traces of Co, Zn, Ag, Mn, Bi and Eu present in the structural materials result in important amounts of activated decommissioning waste. Minimal quantities of galvanised steel or stainless steel and low Ag content in contacts or solder will guarantee lower specific activity at the end of the operational life of the facility. Replacement by low Co-steel, Al or plastic materials for structural materials is recommended. Attention has to be paid to the type of corrosion protection applied to steel: applications of paints or coatings containing metal traces, which can be activated hence creating possible surface contamination, have to be avoided.

A further possibility in limiting the volumes of activated waste is to reduce the space in which the neutrons can interact with infrastructure and shielding. This can be realised by placing local shielding around the targets or other neutron sources.

If prevention of activation of the main walls of the shielding is not possible, judicious modular construction will limit the volume of activated material and ease the distinction and segregation of nuclear from industrial decommissioning rubble. The main idea is to build all walls (except perhaps the ceiling) in a minimum of two layers. The innermost layer, closest to the neutron-generating target, will act as neutron absorber with little structural function. This part of the wall should ideally be built of modular blocks thick enough to reduce the neutron flux to levels below the critical ones for inducing specific activation higher than the clearance levels. Dimensions should be such that easy removal at dismantling and compliance with regulations for disposal of nuclear waste will be met. The second, mainly structural layer can be executed either in ordinary, massive, reinforced concrete, either in modular blocks. The thickness of this wall can already be reduced in comparison to normal shielding as the first layer, needed for radiation protection, will ensure an important part of the dose reduction. Only γ-rays and neutron beams strongly reduced in energy and flux can reach this wall.

CONCLUSIONS AND RECOMMENDATIONS

The decommissioning of particle accelerators can be considered to be a technical operation without any particular difficulty. In fact, the low level of radioactivity within the structures of the equipment and the surrounding shielding material allows the use of hands-on techniques in a radiological protected area similar to that during maintenance and repairs. No strengthened protective measures are required for decommissioning, except for avoiding the spread of activity during dismantling operations (e.g. use of explosive techniques, special attention for ^{65}Zn).

The amounts of radioactive waste can however be considerable due to the activation of significant masses of the shielding structures. Consequently, the philosophy of clearance and the levels for clearance largely influence the costs of the decommissioning programme. The sensitivity to the various philosophies and differences in legal and/or recommended values for clearance, has been highlighted in this report trough the results of the evaluations. Furthermore, the important differences in labour costs and waste management costs within European countries also influence the results of the economic analysis.

Finally, it needs to be highlighted that the decommissioning costs for accelerators amount to about 50 to 100% of the today's investment costs for such accelerators. This fact has probably been undervalued by the operators of accelerators, which can be illustrated by the results of the questionnaire. More attention for building up financial provisions for the later decommissioning seems to be necessary, even if technological development reduces the beam losses in new facilities, reducing activation in the surrounding biological shielding.

Following recommendations can be made:

A. Recommendations for management of existing and future accelerator facilities:

1. Awareness in due time of the problem has to be encouraged so that all information needed for future dismantling is recorded and documented by the operator.
2. National competent authorities should agree in an international framework on the waste management at decommissioning accelerators. Well-defined quantitative clearance levels accompanied by clear conditions and constraints for application are recommended. The conditions for waste management and disposal for all types off materials should help to respect basic radiation protection and environmental criteria.
3. Regulatory authorities should enforce preparation of preliminary dismantling plans in an early stage of the accelerator operation.
4. Mechanisms and funds for provisioning decommissioning costs should be foreseen and enforced.

B. Recommendations for management, operators and constructors of new accelerator facilities

5. The optimisation in the use of nuclear techniques and life cycle analysis of concepts of installations and materials used can decrease the future impact of the waste production.
6. As prevention is always better than curing: adapted technical solutions are recommended for the reduction of source terms, with the implementation of modular construction concepts to limit the waste volumes and costs.

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Measurement of natural gamma radiation in Belgium by means of high resolution in-situ spectrometry

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INTRODUCTION

The availability during the last decade of portable large intrinsic germanium detectors has greatly enhanced the application field of in-situ γ–spectrometry. This technique (described by Beck et al. (1), Miller et al. (2, 3) and in the ICRU report 53 (4)) was initially used for measurements of the fallout from atmospheric nuclear bomb testing.
It has been particularly useful in the follow up of the consequences of the Chernobyl accident in Europe. In the period 1995-1997, we have performed a series of in-situ γ–spectrometry measurements on 60 locations distributed across the Belgian territory. The first aim of this campaign was to make a survey of the ^{137}Cs contamination in Belgium (5,6), but careful analysis of the spectra also produced a nice set of data concerning the natural radioactivity of the ^{40}K, ^{238}U-series and ^{232}Th-series. Our results are compared with data obtained from an airborne survey using sodium-iodide crystals.

EXPERIMENTAL PROCEDURE

The in-situ measurements were performed with three different measuring chains, two from the Ghent University (RUG) and the other of the SCK•CEN Mol. During the first measuring campaign in 1995, the RUG detector is a 20.2 % p-type (energy resolution 1.7 keV) with a normal 30 liter dewar, coupled to on S2011 amplifier and a S20 multichannel analyzer. The SCK chain uses an n-type HPGe coaxial detector (resolution 1.83 keV and efficiency 10%) mounted on a small multi-attitude cryostat. Both spectrometers were calibrated independently and intercompared during simultaneous in-situ measurements at locations in Mol and Tihange. All results are in good agreement within a total experimental error of about 15%. In 1996 and 1997 a series of complementary measurements with a new larger detector (34% efficiency and 1.75 keV resolution with a 7.5 liter multi-attitude cryostat) and a new measuring chain with an integrated InSpector module (Canberra Ind.) were carried out.
The detectors are always placed 1 meter above ground level in the center of a large, undisturbed grassland. Figure 1 shows pictures of the old (1995, left) and new (1997, right) setup in field conditions.

Fig. 1 : Photographs during measurements with both RUG-detectors

A typical spectrum with the large detector in a region with average concentrations of both natural and man-made (^{137}Cs) isotopes is shown in fig. 2 (measuring time 3000s). The energies of the most prominent γ-lines from the decay of the uranium and thorium series and the potassium-line are indicated. In the activity concentration calculations we assume a homogeneous distribution of the natural isotopes in the soil, corresponding to a relaxation parameter value of $\alpha = 0\ m^{-1}$. For the calculations of the ^{137}Cs concentration we use a depth profile with relaxation parameter $\alpha = 20\ m^{-1}$ (relaxation length 5 cm, which is applicable to aged fallout (table 3.5 in (4)). The ^{137}Cs concentration of 2230 Bq/m² in Chimay is a typical value for our country (5,6).

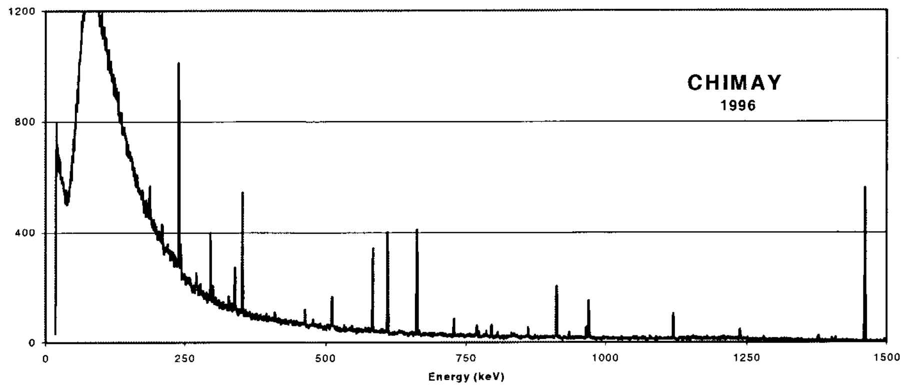

Fig. 2: Typical in-situ γ-spectrum, measured near Chimay in the south of Belgium

RESULTS

The results for the Chimay-site after analysis with CI Genie2K software are summarized in the following table:

^{40}K	1460.7				keV
	260				Bq/kg
^{238}U-series	**average**	242.0	295.2	351.9	keV
^{214}Pb	**26**	29.4	21.9	27.2	Bq/kg
		609.3	1120.3		
^{214}Bi	**25**	25.1	24.2		Bq/kg
^{232}Th-series	**average**	238.6			keV
^{212}Pb	**30**	30.1			Bq/kg
		338.3	911.2	969.0	keV
^{228}Ac	**27**	25.0	27.4	29.7	Bq/kg
		583.2			keV
^{208}Tl	**26**	26.0			Bq/kg

As one can see, 5 different lines from the U-series and 5 for the Th-series give an unmistakable identification of the isotopes.
The concentrations have errors in the order of 20% and there is a good agreement between the concentrations calculated from different lines and isotopes from both series.
The overall averages 25 Bq/kg for the ^{238}U-series and 28 Bq/kg for the ^{232}Th-series are a good indication for the situation in Belgium.
Contour and post maps for potassium (fig. 3), uranium (fig. 4) and thorium (fig. 5) were created from our database with the Kriging gridding method using SURFER for Windows.
The numbers on the axes are the Lambert coordinates, a coordinate system widely used in Belgium; the country is situated between 49.5 and 51.5 N and 2.5 and 5.5 E.

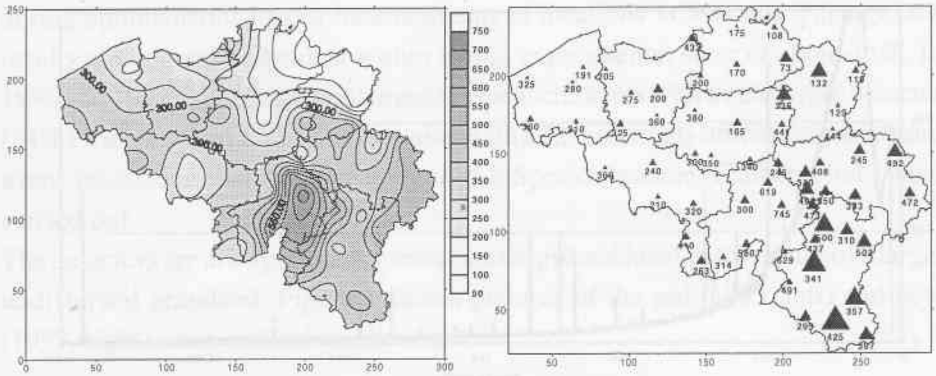

Fig. 3: Contour and post map for ^{40}K. The values are in Bq/kg

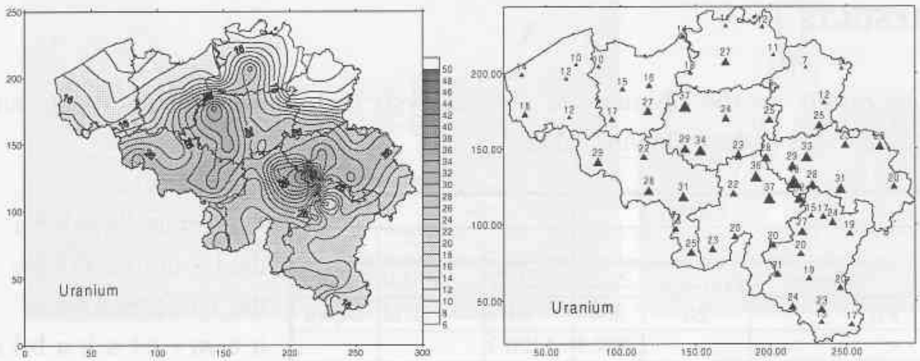

Fig. 4: Contour and post map for uranium. The values are in Bq/kg

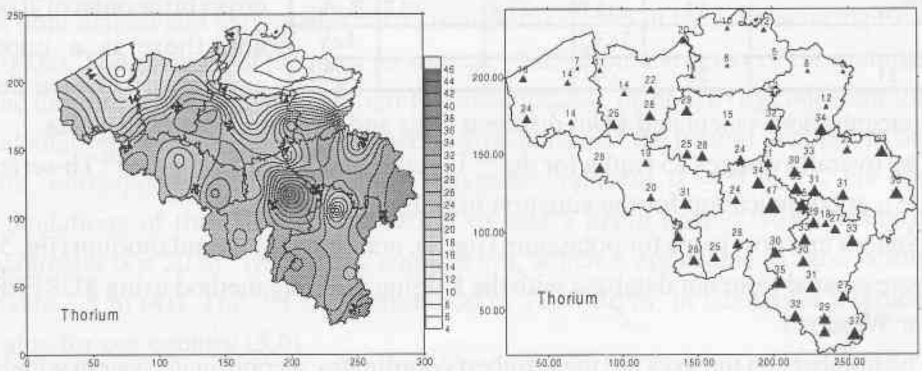

Fig. 5: Contour and post map for thorium. The values are in Bq/kg

DISCUSSION

As one can see on the maps, even a limited number of in-situ measuring points (60) gives a satisfactory overview of the isotope concentrations. The in-situ technique with a detector 1 meter above ground has the intrinsic advantage that a global value for a circle with 25 meter diameter around the detector position is obtained. As we can assume a relative uniform distribution of natural isotopes in the soil, even a limited number of 60 data points can provide sufficient information for map creation.
The reliability of the data is very good due to the use of high resolution gamma spectroscopy. The results are in agreement with geological data obtained from γ-spectrometry on soil samples. For example, in the Ghent region the data for ^{238}U, ^{232}Th and ^{40}K are respectively 14, 15 and 275 Bq/kg from the in-situ measurements and 16, 14 and 290 Bq/kg as an average value for 3 sets of soil samples .
The differences between the north-western part an the central and south-east part of the country can be explained by geological factors: in the Ardennes we find more rocky soil layers containing uranium and potassium.
We have also compared our results with the data of an airborne survey carried out in 1994 for the Ministry of Economic Affairs in Belgium (7). An airplane flying at 120 m above the ground level was equipped with a large scintillation detector and a 256-channel analyzer. Energy windows were used to separate the counts from ^{40}K (1.35-1.57 MeV), ^{238}U-series (1.63-1.89 MeV), and ^{232}Th-series (2.42-2.82 MeV). Corrections for cosmic and aircraft background were carried out.
The values on the airborne survey maps are in counts/s (cps), with approximate equivalence coefficients for the conversion to Bq/kg. The correspondence between our data and the airborne survey measurements is in general reasonable, taking into account the completely different methodology and detection systems involved. For example, in the Ghent region the airborne survey value of 100 cps for ^{40}K can be converted to 270 Bq/kg which value is very close to the 275 Bq/kg from the in-situ measurements.
Some discrepancies, especially in the uranium measurements, can probably be explained by weather effects on the gaseous radon in the ^{238}U decay scheme. The airborne survey is also sensitive to interferences with man-made sources of enhanced “natural” radioactivity, like the uranium containing stockpiles and waste storage grounds from some industries (phosphate and fertilizers factories, gypsum manufacturers, etc.).

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A model to establish the monetary value of the man-sievert for public exposure

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INTRODUCTION

In order to select, among the range of possible protective actions, those which are consistent with the available resources and which ensure that the risks are distributed satisfactorily between individuals, the use of economic analysis (through cost-benefit analysis) was introduced in the early 1970s by the ICRP (1, 2, 3). In this perspective, the monetary value of the man-sievert appears to be a central element for the implementation of the optimisation of the radiation protection. For occupational exposures, a model to determine the monetary value of the man-sievert has been developed (4) and some operators of nuclear installations and authorities have gradually adopted monetary values as a decision-aiding tool for the evaluation of major protective actions. Recent reflections have pointed out the needs for differentiating the monetary values of the man-sievert according to the exposure situations under consideration (5). The aim of this paper is to propose a model for the determination of the monetary values to be applied for the evaluation of protective actions related to public exposures.

ROLE AND EVALUATION OF THE MONETARY VALUE OF THE MAN-SIEVERT

The aim of optimisation of radiation protection is to obtain an efficient allocation of protection resources. The efficiency criterion does not rely solely on the financial return on investments, but also on ethical and social dimensions related to the acceptability of risk according to the various exposure situations. As such, the monetary value of the man-sievert is a key element for the implementation of optimisation allowing to take into account the detriment as well as the benefits expected from activities in the allocation of protection resources. This allocation has notably to take into account the risk transfers between populations.

From the economic point of view, the monetary value of the man-sievert can be seen as a function reflecting the "utility" (or disutility) or individual and collective preferences associated with the level of exposures and the specific exposure situation. For this purpose, it has to integrate several dimensions:

- One dimension, which is independent of the exposure situation, is related to the potential health effects associated with the level of exposure. These health effects consist mainly in delayed cancers (fatal and non fatal) and severe hereditary effects. Within the economic theory, several methods already exist for the valuation of health detriment (Quality Adjusted Life Year, Value of Statistical Life based on Willingness to Pay Approach or Human Capital Theory). Therefore, the knowledge of the dose-effect relationship allows to determine the monetary value of the health detriment, which forms one component of the monetary value of the man-sievert. It has to be noted that the health detriment, when it occurs, is the same for public, workers or future generation individuals whatever exposure situation is considered.

- Besides the health detriment dimension, it is necessary to introduce social and equity dimensions which should allow to reflect the specificity of the various exposure situations: characteristics of the situation, distribution of individual exposures, individual and social risk perception, weighting of probabilities and consequences for potential exposure, aversion towards uncertainty when dealing with future generations etc…

METHODOLOGY DEVELOPED FOR THE EVALUATION OF PUBLIC EXPOSURES

Differences between public exposures and workers exposures

In the case of occupational exposure, a model for defining the monetary values of the man-sievert was developed a few years ago and has favour the adoption of monetary values by some operators of nuclear installations and authorities. Although the reference to the monetary value of the man-sievert is observed in a number of decisions related to the protection of workers, there was only a few discussions for public exposures.

It is notably considered that for these latter the monetary value of the man-sievert should be different.

In this perspective, the main differences between an individual exposure for a member of the public and for a worker have been considered in terms of their influence on the willingness to pay for a reduction of exposure (reduction of the probability of radiation induced cancer). After an analysis of a number of differences (level of initial exposure, age at exposure, compensation system…), the possibility to compensate the workers in case of radiation induced cancer was considered as the major difference between workers and public exposures from the economic point of view.

In practice, compensation systems have been implemented for the workers exposed to ionising radiations and having developed a cancer. In the case of public exposure, such systems do not exist, mainly due to the absence of a permanent individual monitoring of exposures and to the low level of exposures (6).

Theoretical model for the monetary value of the man-sievert for public exposures

In order to determine the monetary value of the man-sievert applicable for public exposure, the theoretical approach adopted consists in evaluating the willingness to pay for a reduction of exposure in case of public exposure compared with the same reduction for occupational exposure (7). Based on the expected utility approach, the following parameters have been considered in the model:

- the total wealth of an individual in the absence of radiation induced cancer;
- the loss associated with the occurrence of a radiation induced cancer (this parameter being less important when a compensation exists);
- the probability of occurrence of a radiation induced cancer according to the exposure level;
- the reduction of exposure.

In order to characterise the utility function reflecting the preferences of the individual, a relative risk aversion coefficient was considered. This relative risk aversion coefficient reflects the attitude of individuals towards risk.

This model leads to the conclusion that there is in general a negative influence of the level of the compensation on the value of the willingness to pay. Therefore, given the fact that the member of the public cannot be compensated if they declare a radiation induced cancer, the willingness to pay to reduce their probability of cancer (i.e. to reduce their level of exposure) should be higher than the willingness to pay applied to reduce the probability of cancer for the workers.

PROPOSED MONETARY VALUES FOR PUBLIC EXPOSURES

In order to determine a monetary value of the man-sievert for public exposure, a numerical application of the theoretical model has been proposed. For this purpose, the following parameters have been characterised: range of probabilities of radiation induced cancer for public and workers exposures, reduction of exposures, level of wealth and compensation system. On this basis, the evolution of the willingness to pay has been analysed for different levels of relative risk aversion (ranging from 1 to 3).

Range of probabilities for public and workers

The value of the initial probability of radiation induced cancer for public and workers has been evaluated on the basis of the dose-effect relationship associated with ionising radiation (8). The situations considered for assessing these probabilities refer to individual exposure levels reflecting current situations.

- If we assume a member of the public exposed at 0.1 mSv/year during 75 years, his lifetime dose is equal to 7.5 mSv, and, applying the dose-effect relationship, his lifetime risk (probability of occurrence of a radiation induced fatal cancer) is equal to 4. 10^{-4}.

- If we assume a worker exposed at 5 mSv/year from age 18 to 65 years, his lifetime dose is equal to 240 mSv, and his lifetime risk is equal to 10^{-2}.

In both situations, the initial probability of a radiation induced cancer, considered in the numerical application, corresponds to the lifetime risk.

Reduction of the exposure level

For comparing the willingness to pay for a reduction of probabilities of radiation induced cancer for workers and public exposures, a reduction of exposure leading to a decrease in probability equal to 10^{-4} has been considered. Therefore, the probabilities after the reduction of the exposure level are respectively for public and workers: $3 \cdot 10^{-4}$ and $99 \cdot 10^{-4}$.

Level of wealth and loss of wealth

For evaluating the utility of the individual in the different exposure situations, the initial level of wealth, expressed in monetary terms, is considered. This wealth is supposed to be the same for a member of the public and a worker: 915 KEUROS (rounded value). It is based on two components: the monetary value of life (about 850 KEUROS) and the average individual financial wealth (estimated to 65 KEUROS).

The loss of wealth in case of occurrence of a radiation induced cancer is evaluated on the basis of its associated loss of life expectancy (GDP/capita x loss of life expectancy due to a radiation induced cancer): 305 KEUROS (rounded value).

Compensation system for occupational radiation induced cancer

Three situations have been considered for the levels of compensation for the workers: no compensation (just for the sake of comparison with the public situation), compensation of 50 % of the loss of wealth or compensation of 75 % of the loss of wealth.

In fact, one may consider that the compensation systems already implemented in the case of occupational diseases, and more specifically in the case of occupational cancers, generally cover expenses in the range of 50 to 75% of the loss of wealth.

The main results

From the numerical application, a ratio between the willingness to pay for a reduction of probability of radiation induced cancer for a member of the public and the one for the workers has been calculated. Depending on the level of compensation for workers and on the relative risk aversion coefficient for the utility function, the willingness to pay for a reduction of probability of developing a radiation induced cancer for a member of the public should be between 2 and 6 times higher than that of a worker.

Figure 1 summarises the results obtained for the different values of the relative risk aversion coefficient.

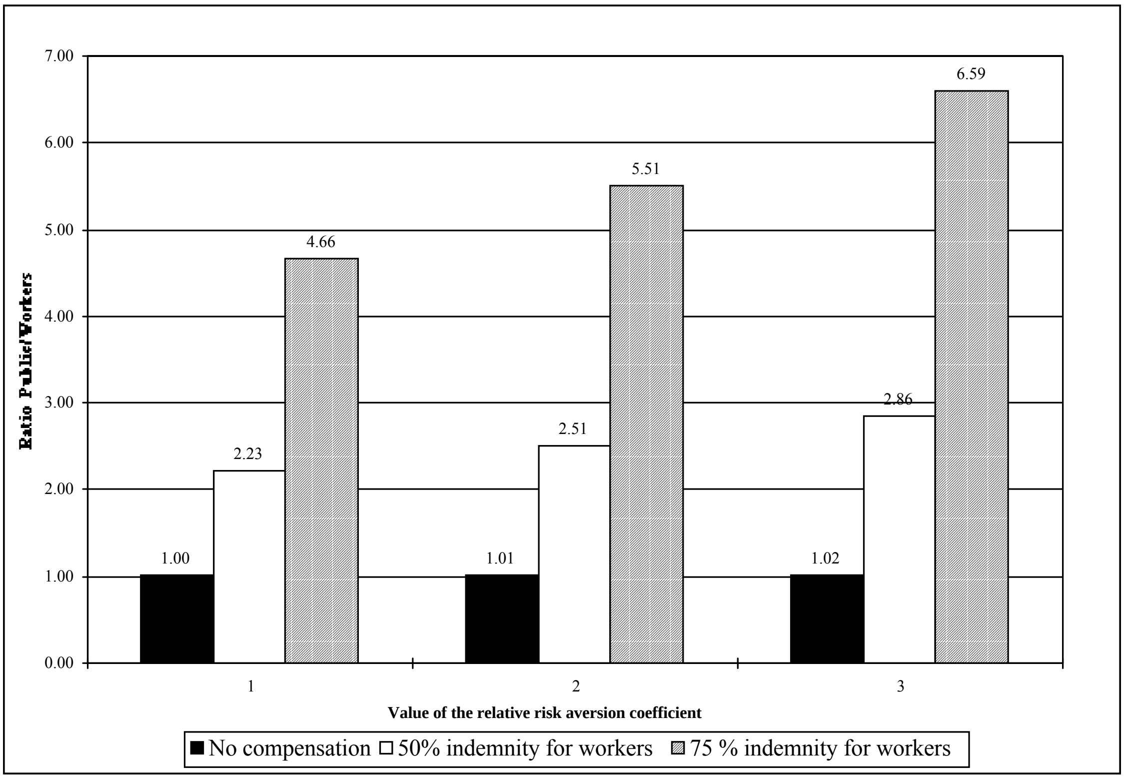

Figure 1. **Ratio of the WTP for a small reduction of probability of occurrence of a radiation induced cancer according to increasing values of the relative risk aversion coefficient**

On this figure, one should notice that in the absence of a compensation system for workers, the willingness to pay for a reduction of probability of radiation induced cancer is similar between workers and public.

Application to the monetary value of the man-sievert

On the basis of the human capital approach, the monetary value of the health effects is given by valuing one year of loss of life expectancy with the Gross Domestic Product per capita per year. According to the dose-effect relationship for ionising radiation, it leads to a monetary value of the radiation health effects per man-sievert for the public equal to: 25 KEUROS/man.Sv.

Furthermore, a recent survey conducted in France to evaluate the willingness to pay to reduce the probability of a radiation induced cancer in case of occupational exposure in nuclear power plants allowed to obtained a monetary value of the human life in case of death by cancer equal to 450 KEUROS (9). According to this value, and using the probability of occurrence of a radiation induced health effect for the public presented above, the following monetary value of the man-sievert is obtained: 33 KEUROS/man.Sv.

Adopting a relative risk aversion coefficient between 1 and 3 for the utility function, and assuming that usually the compensation for a radiation induced cancer is in the range of 50 % to 75%, the following set of multiplying coefficients is proposed in order to derive the monetary values of the man-sievert for public exposures from the values related to occupational exposures: 2; 4 and 6.

Table 1 gives the resulting monetary values of the man-sievert for public exposures when the different multiplying coefficients have been applied.

Table 1. Proposal for the monetary values of the man-sievert in case of public exposure

Basic monetary value of the man-sievert	Multiplying coefficient		
	2	4	6
25 KEUROS/man.Sv	50 KEUROS/man.Sv	100 KEUROS/man.Sv	150 KEUROS/man.Sv
33 KEUROS/man.Sv	66 KEUROS/man.Sv	132 KEUROS/man.Sv	198 KEUROS/man.Sv

The resulting monetary value of the man-sievert to be applied in optimisation studies for the reduction of public exposures ranges from 50 KEUROS/man.Sv to 200 KEUROS/man.Sv.

CONCLUSION

The theoretical model used to evaluate the monetary value of the man-sievert suggests a higher willingness to pay to reduce the probability of occurrence of a radiation induced fatal cancer for the public than for workers. This difference is mainly related to the non existence of a compensation system for members of the public. For worker exposure, compensation systems are implemented in case of recognition of occupational diseases, including cancer potentially induced by radiations. The numerical application of the model shows that depending on the level of compensation for workers and on the relative risk aversion coefficient adopted for the utility function, the willingness to pay for a reduction of probability of developing a radiation induced cancer for a member of the public should be between 2 and 6 times higher than that of a worker. The resulting monetary value of the man-sievert to be applied in optimisation studies for the reduction of public exposures is ranged between 50 KEUROS/man.Sv and 200 KEUROS/man.Sv.

ACKNOWLEDGEMENT

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Data assimilation and monitoring strategies during the early phase of a severe accident

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A coherent and effective off-site emergency management is a key element in case of a radiological emergency to protect the public and the environment and forms the last of three pillars of nuclear safety. For this reason the novel European RODOS (Real-time On-line Decision Support System) has been developed, with as main goal, to assist the decision makers during all the stages of a nuclear emergency in their tasks. A lot of research within the RODOS project has been and is still being devoted to data assimilation. The aims of data assimilation are three-fold: 1) to acquire in an integrated way strategic information, such as subjective expert judgment, data, measurements and model predictions, 2) to perform the necessary analyses to allow to construct a comprehensive picture of the required radiological contamination fields and 3) to use the thus built-up knowledge to propose suitable sets of countermeasures to the decision maker.

For nuclear installations located in densely populated areas, the early phase represents a challenge as decisions on adequate interventions have to be taken under high time pressure, under stress conditions and using information which is not comprehensive or even conflicting. In this paper an overview is given of work being done at SCK•CEN within the RODOS project related to the field of data assimilation during the early phase.

The main idea has been to acquire enough information to estimate the source term (real-one or expected one) and to feed it in a suitable dose/dispersion model, allowing to perform the necessary dose assessments in time and space. The module OSSTERM (Off-Site Source TERM) has been developed to acquire the best view as possible on the source term in real-time based on the assimilation of off-site monitoring data (basically fence monitoring gamma dose rates) and model predictions. The model predictions are initially fed by the initial (pre-release) estimation of the main characteristics of the release. First validation results using simulated datasets are encouraging. The need for further research to make the methodology fully operational will be discussed.

Some difficulties and limitations of data assimilation during the early phase are pointed out, esp. the importance of better monitoring strategies. In particular, it is shown that most actual off-site monitoring programmes could be much better tailored towards the needs of data assimilation.

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Cancer Mortality among Nuclear Workers in Belgium

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INTRODUCTION

Animal experiments and epidemiological data in humans have provided evidence that high doses of ionising radiations can induce cancer (1). However, cancer risk estimates available so far , are based mainly on studies about short term exposure (acute) to high radiation dose, e.g. among radiotherapy patients and atomic bomb survivors (2-7). Cancer risk estimates after chronic exposure to low doses of ionising radiation are, in most cases, obtained from extrapolation to low doses of the data observed after high-dose exposure. Radiation protection regulations and current practices are based on these extrapolations (8). Such types of extrapolation imply of course assumptions, in particular concerning the shape of the initial part of the dose-effect relationship. Therefore, cancer risks after low radiation exposure are difficult to evaluate and current estimates remain controversial.

Because it is important to know if the present radiation protection measures are adequate and safe to protect the public as well as the radiation workers, there is a need to measure the cancer risk directly on populations chronically exposed to low doses, such as e.g. radiation workers. The present report is dealing with nuclear workers: this group has been selected because these workers are occupationally exposed to low doses of radiation during several years, and individual data on exposure are available.

Most of the studies that have been done in nuclear workers investigate cause specific mortality rates in workers of different dose categories. They include workers of several types of installations such as nuclear research centers, reprocessing plants, power plants, military installations. But they have some common characteristics as well: most of the workers are exposed to low doses of external radiation (X and gamma radiation) and individual yearly dosimetry data have been registered and kept since the years 1940-50. Only a minority of workers has been exposed to neutrons, or has been at considerable risk for internal contamination.

When total mortality (all causes of death) or cancer mortality (all types of cancer) are compared for the nuclear workers and the general population, only few studies find a significant higher mortality. No specific cancer type is consistently associated with cumulated dose. On the contrary, many studies describe a significantly lower mortality than in the general population, probably influenced by the so called "healthy worker effect", a selection bias. When looking for association with cumulated dose, some positive trends have been found, some of them significant. Leukemia mortality is most frequently associated with dose (9-13). Concerning internal contamination, most of the studies only flag the workers monitored for internal contamination, but have no individual data. Some research has however been done on particular radionuclides, such as plutonium (14).

The many individual studies among nuclear workers done so far often lack statistical power because the effect of low doses of radiation on cancer incidence appears small. Therefore, the International Agency for Research on Cancer (IARC/WHO) in Lyon, France, decided to pool available data. The combined data of several already published studies from Canada, the UK and the USA were analysed to obtain more precise estimates of cancer risk. No association was found between cumulative dose and total cancer mortality. An Excess Relative Risk of 2,2 per sievert cumulated dose for leukemia mortality was observed, but still within a broad confidence interval (0,1-5,7). Therefore it is concluded that so far "there is no evidence that the current radiation protection measures are appreciably in error" (ref. 15-16).

Following this combined analysis, IARC set up a multicenter study, coordinated by Dr. E. Cardis, to pool data of 14 different countries. Belgium is one of the participants.

OBJECTIVES

To investigate cause-specific mortality with emphasis on cancer in Belgian nuclear workers, the Nuclear Research Center (SCK.CEN) in Mol set up a retrospective cohort study in the five largest nuclear facilities in Belgium. Mortality among radiation workers is studied in relation to their occupational exposure to ionising radiation. An additional objective of this epidemiological research is to make recommendations to facilitate the conduct of retrospective cohort studies in Belgium. As mentioned above, this study is part of the “International Collaborative Study on Cancer Risk among Radiation Workers”(IARC/WHO).

METHODS

To reach these objectives, cause specific mortality is studied retrospectively (1953 – 1994) in a cohort of workers from the following 5 nuclear facilities in Belgium:

- the Nuclear Research Center in Mol : follow up period 1953-1994
- Belgoprocess (BP), treating nuclear waste in Mol: follow up period 1985-1994
- Belgonucleaire (BN), a MOX fuel production facility in Dessel: follow up period 1957-1994
- Electrabel NV, the nuclear power plants of Doel (KCD) and of Tihange (CNT) : follow up period 1974-1994

During the period 1953- 1994, all workers registered for at least one month in the personnel register of one of the participating facilities, were included in the study (n= 7361). All staff members thus meet the inclusion criteria, but no contract workers. A pilot study showed that too many of the latter would have been "lost to follow up".

For each worker, we collected the following set of data, using different information sources:

- demographic data (identification, sex, age..): mainly from the personnel registers
- occupational history and jobdecription: from personnel registers, medical records, Ministry of Labour
- exposure to ionising radiation : mainly extracted from the dosimetry records available in the facilities (annual effective dose, flaggings for neutron exposure, internal contamination risks) and from medical records. To assess the transfer doses, the National Radiation Registry kept by the Ministry of Labour was consulted. Collection of dosimetry data before 1973 was essentially done manualy from paper files.
- vital status on 31-12-94: assessed through the National Population Registry & local authorities, pension funds and the facilities social services
- causes of death , extracted from the death certificates : information (underlying and direct causes of death, ICD9 encoding) was obtained from the National Institute of Statistics, the French and Flemish Community. Written informed consent of next-of-kin was required to obtain information from the death certificates (home visits). Unfortunately, in Belgium individual death certificates are no longer available before 1969. For the earlier period , only family reported causes of death - obtained through home interviews – are available.
- smoking habits: a smoking survey was done among current workers.

As a first approach, Standardised Mortality Ratio's (SMR) were calculated for underlying causes of death. The general Belgian population was choosen as the reference population, because workers originated from all over Belgium. SMR calculations were performed in collaboration with the Epidemiology Department of the University of Maastricht. The PETO programme was used. Data were corrected for unknown causes of death (assumption: no relation between exposure and access to death certificate). Workers were categorised in the following dose categories : "non measurable dose" and "measurable doses"

The cohort we describe in this paper is restricted to the period 1969-1994, and to the workers of SCK.CEN, BN and BP. As we already mentioned, individual death certificates are no longer available for the earlier period, and thus SMR's can only be calculated from 1969 on. Only direct standardisation based on lay reported causes of death is possible for the 1953-1968 study period. The Electrabel workers cohort is not included in this paper because collection of data is still ongoing.

Further analyses of Relative Risks (RR) in different exposure categories (less then 10 mSv, 10-<50 mSv, 50-<100 mSv, 100 mSv and above) will be done in collaboration with the University of Maastricht and the Catholic University of Louvain-La-Neuve. Our data will be pooled with other data collected within the IARC study.

RESULTS

First results of SMR calculations can be presented for SCK.CEN workers for the 1969-1994 period. A total number of 3270 workers were studied during this period. Despite the above mentioned constraints, we obtained a vital status ascertainment of 95%. For underlying cause of death the ascertainment is now 80%.

Available SMR's can be summarised as follows (Table 1) :

- male workers, no measurable dose (n = 785): SMR for all causes of death = 75% (95% C.I.= Confidence Interval: 61-91), SMR for all tumours = 64 % (95% C.I.: 42-93), 2 leukemia deaths were observed, whereas 1 is expected;
- male workers, measurable dose (n = 1785): SMR for all causes of death = 64% (95% C.I.: 56-74), SMR for all tumours = 62 % (95% C.I.: 48 - 80), 2 leukemia deaths were observed, whereas 3 are expected
- female workers, no measurable dose (n = 553): SMR for all causes of death = 94% (95%CI: 63-135), SMR for all tumours = 106 % (95%CI: 55 - 161), no leukemia deaths were observed
- female workers, measurable dose (n = 147): SMR for all causes of death = 100% (95% C.I.: 57-163), SMR for all tumours = 90 % (95% C.I.: 29 - 208), 1 leukemia death was observed.

		OBSERVED*/EXPECTED	SMR (%)	95% C.I.
Non measurable dose,	All causes	101 / 135.2	75	61- 91
male workers (n=785)	All tumours	25.8 / 40.5	64	42 – 93
	Leukemia	2.3 / 1.1	-	-
Measurable doses received,	All causes	202 / 313.4	64	56 – 74
male workers (n=1785)	All tumours	62 / 99.7	62	48 – 80
	Leukemia	2 / 2.7	-	-
Non measurable dose,	All causes	29 / 30.9	94	63 – 135
Femaleworkers (n=553)	All tumours	12.6 / 11.9	106	55 – 161
	Leukemia	0 / 0,4	-	-
Measurable doses received,	All causes	16 / 15.9	100	57 – 163
Female workers (n=147)	All tumours	5 / 5.5	90	29 – 208
	Leukemia	1 / 0.2	-	-

* + correction for number of unknown causes of death

Table 1. Standardised Mortality Ratio's among SCK.CEN nuclear workers 1969-1994.

CONCLUSIONS

In this study performed in Belgium, SMR's in nuclear workers are significantly lower compared to the Belgian general population for all causes of death and for all tumours in males; in female workers mortality does not differ from the general population, but the number are small. No increase in leukemia mortality was observed, but only few leukemias had to be expected and few were actually observed. Other cause specific mortality rates (25 specific cancer sites, cardiovascular & respiratory diseases, external causes) did not reveal any significantly increased SMR among this worker cohort. A "healthy worker effect" (selection bias) may influence this observation. The correlation between cause specific mortality rates and radiation dose is further investigated.

Retrospective collection of data and privacy protection regulations specific to Belgium hampered the conduct of this study, causing labour intensive and time consuming procedures. Written informed consent of next-of-kin is required to obtain information from the death certificates. Before 1969 only family reported causes of death are available. In order to circumvent these constraints, in the present study, novel approaches had to be developed to collect the needed relevant information. Finally this study has drawn the attention on the need to standardise feasible and appropriate procedures to consult the National Population Registry and to obtain reliable information on individual causes of death.

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The International Commission on Radiation Units and Measurements (ICRU): Activities and Future Plans

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BRIEF HISTORICAL OVERVIEW

Mandate of the ICRU

Thirty years after Röntgen's discovery of the x-rays, the First International Congress of Radiology, at a meeting held in London, acknowledged the need for internationally accepted standards for units and measurements of exposure to ionizing radiation. The Congress appointed the *"International X-Ray Unit Committee"*, now the "*International Commission on Radiation Units and Measurements"* (ICRU), to provide scientific underpinnings and to investigate, develop, and establish such quantities, units, and measures pursuant to these goals. While medical applications of x-rays were initially the major impetus, the protection of individuals was included from the beginning. In 1928, at the Second International Congress of Radiology, an international agreement was achieved on the definition of a *unit* of x-ray dose, the röntgen, based on measurements with air-filled ionization chambers. In 1937, the ICRU recommended a definition of the röntgen applicable to x-rays as well as gamma rays.

Over the time, the field of activities of the ICRU expanded considerably to meet new and increasing needs. Today the objectives of the ICRU can be divided into four main areas:
(1) Quantities and Units,
(2) Radiation Protection and Radiation Ecology,
(3) Medical Applications of Ionizing Radiation,
(4) Basic Data for Radiation Interactions.

Recently, the ICRU decided to enter cautiously and progressively into the field of non-ionizing radiation. Here only the first two aspects of the ICRU program will be reviewed.

Initial steps in radiation measurement

The history of radiation measurements, in general, and for radiation protection purposes, in particular, is also the history of a succession of different quantities and, in fact, units. Soon after Röntgen's discovery of x-rays, many of the physical and, indeed, even biological effects induced by x-rays were used for radiation measurement purposes (Menzel and Feinendegen, 1995).

The radiation-induced blackening of photographic film, which Röntgen observed with x-rays, was one of the early methods suggested for measurements and definitions for units of radiation. Although film has never played a significant role in standardization, film dosimetry has been persuasive for personal monitoring of photon exposures and is still the medium of choice for diagnostic radiography.

Colour changes of chemicals, subsequent to x-ray exposures, led to popular units (e.g., the "pastille unit" (1904) for platino-barium cyanide capsules or the "Holzknecht" or "H" unit (1904) for mixtures of potassium chloride and sodium carbonate) for assessing exposure for a considerable period of time. Also, various photoluminescence effects were detected early in the history of radiation measurements, including thermo-luminescence, which Marie Curie reported as early as in 1904 in her doctoral thesis.

However, the most important physical effect for the initial phase of standardization was the ionization effect in gases. The first ionization chambers were built even before 1900. Many proposals were put forward for units based on radiation-induced ionization leading, finally, to the definition of the röntgen.

All originally proposed units were closely related to the underlying measurement principle. The fact that x-rays played a role predominantly in medical applications in that period explains why correlations to induced biological effects were established using "visible" observations, mainly reactions of skin. Not all of them were considered practical; for example, the proposal to use the depilatory effect of x-rays on hair did not find many supporters.

The radiation induced reddening of skin was used to define a unit called skin erythema dose (SED) which was used widely as a reference and was compared to the units based on physical measurements in an attempt to establish dose equivalencies between physical and biological phenomena.

Dosimetry in radiation protection

The development from the röntgen, the first internationally accepted unit of radiation measurements, to the currently practiced system of *quantities and units*, embedded into a general concept of radiation protection, is the result of the work of ICRU often in collaboration with its sister organization, the ICRP. This evolution was governed:

- by scientific progress in the underlying disciplines of radiation physics and radiation biology,
- by the rapid extension of the use of ionizing radiation and radioactive materials in medicine and in the power generation and other industries since the 1950's,
- by the recognition that hereditary damage and radiation carcinogenesis are the main risks associated with low dose exposures. (In addition, of course, non-stochastic effects are observed at high doses),
- and by the intention to generalize the applicability and to improve the scientific rigour and consistency of the definitions.

The objectives of radiation dosimetry are to provide procedures, techniques and concepts for the determination of the "amount" of ionizing radiation that is quantitatively related to the potential biological effect. In radiation protection, dosimetry is thus concerned with the practical aspects of developing internationally accepted concepts, quantities and methods which are suitable for radiation risk assessment, and can be used for controlling exposures and specifying exposure limits for radiation workers and the public.

QUANTITIES AND UNITS

One of the main objectives of the ICRU is to develop a universally accepted set of quantities and units for radiation and its interaction with biological matter. Only with a set of scientifically sound, easily understood, and well implemented quantities and units can ionizing radiation be safely employed to meet society’s needs in medical care and industrial applications.

The ICRU has developed and recommended a set of fundamental quantities and units which has been in wide use for decades and has been vital to the successful exchange of information, and comparison of results.

However the field is not static and the expanding uses of radiation and radiation-producing processes demand further development and elaboration of quantities which, in turn, implies development and understanding of new concepts that will facilitate meeting the new needs.

The ICRU organization includes a standing committee on "Quantities and Units". As new concepts, instruments, and techniques emerge from the radiation science community, this committee serves to refine the definitions of quantities and units and include and expand them in response to these involving needs.

Development of concepts, quantities and units

Several milestones can be identified in the development of concepts, quantities and units for dosimetry in radiation protection:

- in 1953, at the 7th International Congress of Radiology in Copenhagen, the ICRU introduced the *absorbed dose* in a irradiated material, by any type of ionizing radiation, as the fundamental *quantity* correlated to the induced biological effect. The special *unit* introduced was the rad; now this special unit in the "Système International" (SI) is the gray (Gy).

- in 1962, the ICRU introduced -for radiation protection purposes- the quantity dose equivalent as a product of the absorbed dose and various modifying factors, the most important of which is the quality factor. This factor accounts for differences in the relative biological effectiveness of different types of ionizing radiations atlow doses. The special unit introduced was the rem; now the SI-based special unit is the sievert (Sv).

- in 1977, the ICRP introduced (ICRP Report 26) the effective dose equivalent, based on the dose equivalent in various organs of an individual and the weighted sum of these, as a limiting quantity for all types of exposure. In 1991, the ICRP modified its approach and introduced effective dose.
- in 1985, the ICRU (ICRU Report 39) introduced operational quantities for the specification of dose equivalent, for area and individual monitoring in the case of external radiation sources.

Physical quantities, protection quantities and operational quantities

Determination of radiation protection quantities often involves significant uncertainties. In addition, a variety of approximations must be made to relate physical measurements to the biological effects caused by radiation. Therefore, although the requirements for accuracy in radiation protection measurements may not always be high, it is essential that the quantities employed be defined unambiguously, and that any approximations be clearly stated.

The ICRU's currently recommended system of radiation protection quantities and units fulfills these requirements and has beenadopted by most national and international regulatory bodies. The system can be described as a hierarchy of quantities composed of physical quantities (including fluence, kerma and absorbed dose), protection quantities (effective dose, organ doses) and operational quantities (ambient dose equivalent, directional dose equivalent, and personal dose equivalent).

ICRU Report 51, *Quantities and Units in Radiation Protection Dosimetry* (1993) summarizes all current definitions and takes account of the new formulations of protection quantities contained in ICRP Publication 60 (1990). More recently, ICRU Report 60 (1998), "Fundamental Quantities and Units for Ionizing Radiations", provides specifics regarding all quantities and units underlying physical determinations ultimately used in protection dosimetry. The ICRU will expand the definitions into area of radioecology and publish a new report, Report 65, "Quantities, Units and Terms in Radioecology".

To support and promulgate these physical quantities, numerous national standards (as well as secondary standards) laboratories maintain a range of well-defined radiation standards. The ICRU works closely with these organizations to ensure that optimal execution of the standards base is maintained. These laboratories provide a network of services for instrument and dosimeter calibration in terms of operational and base units. ICRU Report 64, "*Dosimetry of High-Energy Photon Beams Based on Standards of Absorbed Dose to Water"* (in press), provides information on current methods used in primary standards laboratories.

The protection quantities, introduced by the ICRP, are conceived to be proportional to the potential radiation risk for almost all types of radiations. They were introduced mainly for the purpose of exposure control and thus risk limitation.

Operational quantities are defined in a 30 cm diameter sphere of ICRU tissue and are conceived to provide an (almost always conservative) estimate for the relevant protection quantities. Personal and environmental monitoring with instruments calibrated using ICRU-defined operational quantities permits the control of exposure as well as the assessment of the overall exposure.

The numerical relationship between physical quantities such as particle fluence and air kerma, and the protection and operational quantities can be determined using the computational methods of numerical dosimetry, anthropomorphic phantoms for protection quantities, and the ICRU sphere for operational quantities.

ICRU Report 57: *Conversion Coefficients for use in Radiological Protection against External Radiation* is the product of collaboration between the ICRU and the ICRP. The report was prepared by a joint task group of the two Commissions, and underwent separate reviews by each of the Commissions and has been published in the report series of both Commissions.

The Report was published as ICRP Publication 74 in the beginning of 1997 and as ICRU Report 57 early in 1998. It provides an extensive and authoritative set of data-linking field quantities, operational quantities, and protection quantities in a way that will be of help to those working in radiation protection for external exposure. Conversion coefficients are provided for idealized irradiation geometries, monoenergetic radiations in anthropomorphic phantoms (mathematical models) and measurement phantoms. Publication of ICRU Report 57 is timely in view of the European Commission (EC) Directive revising the Basic Safety Standards adopted by the European Council in May 1996 and the publication of the International Basic Safety Standards published by the IAEA.

RECOMMENDATIONS FOR RADIATION PROTECTION MEASUREMENTS

Besides involvement in the development of concepts and quantities, the ICRU has always played an important role in providing guidance for radiation protection measurements.

An early example is ICRU Report 20, "*Radiation Protection Instrumentation and its Application"* (1970). ICRU Report 36, "*Microdosimetry"* was published in 1983. Later on, the ICRU provided guidance for the determination of operational quantities in three reports: Report 39 "*Determination of Dose Equivalents Resulting from External Radiation Sources"* (1985), Report 43 "*Determination of Dose Equivalents from External Radiation Sources* (Part 2)"(1988) and Report 47 "*Measurement of Dose Equivalents from External Photon and Electron Radiations"* (1992). A report on " *Determination of Operational Quantities for Neutrons" is in preparation.*

In addition to issues concerning dosimetry for monitoring external irradiations, the ICRU is addressing topics in the overall area of radiation and radioactivity measurements for protection purposes. Examples are, the ICRU Report 52, "*Particle Counting in Radioactivity Measurements",* was published in 1994 and Report 53, "*Gamma Ray Spectrometry in the Environment"* (1995). The latter was prepared in recognition of the fact that methods for quickly assessing radionuclides in the environment have become increasingly important, particularly in the context of accidental releases from nuclear facilities.

Gamma-ray spectrometry, based on the measurement of the energy distribution of the photon fluence, is used for the determination of activity levels in the ground or in air and of radionuclide-specific dose quantities. It is also applied to the control of planned releases, in dose reconstruction and environmental remediation projects and in the search for radioactive sources in the environment.

Report 56, *Dosimetry of External Beta Rays for Radiation Protection*, published in 1997, recognizes that the general aim of beta-ray dosimetry in radiation protection is to provide dosimetric information that will help in keeping any harmful effect of beta rays within acceptable limits and that, in the event of serious over-exposure, will assist medical treatment and hence prognosis.

Other reports on specific measurement techniques are in preparation, including "*In vivo Determination of Body Content of Radionuclides"* and in the area of radiation protection of patients, on "*Dosimetric Procedures in Diagnostic Radiology".*

CONCLUSION

The ICRU played a decisive role in the development of concepts and quantities for radiation dosimetry in radiation protection for 70 years.

The current system of quantities for radiation protection dosimetry meets the requirements of scientific rigour and practical applicability for monitoring in almost the entire range of exposure conditions of workers and the public.

In the field of quantities and units, one of the main contributions of the ICRU is the introduction of the concept of absorbed dose to quantify "the amount of radiation". The quantity absorbed dose (with its special unit gray) is found to be most useful for the majority of radiation applications above all in radiation medicine, and is universally accepted.

However, some limitations in the concept of absorbed dose have been identified in particular when the conditions for the "averaging procedure" implied in the concept are no longer met Examples of such cases include: administration of low activities of radionuclides emitting weakly penetrating radiation or short-ranged particles in nuclear medicine diagnostic studies, situations of a few particle tracks (protons, ions,..) in radiation protection, therapy with unsealed sources, BNCT, etc.

The ICRU is considering this issue very carefully and is preparing a report. This report ("Dose specification in nuclear medicine") will be limited, for the moment, to administration of low activities of radionuclides for diagnostic and therapeutic purposes in nuclear medicine. The report specifically deals with inhomogeneous distributions of radionuclides as well as tissues inhomogeneous in density and composition. Low doses and low dose rates are inherent to such situations. Moreover, energy deposition, i.e. imparted energy, is considered at a sub-millimeter level. This report provides a link between macroscopic- and microscopic dosimetry. The potential implications for radiation protection are apparent.

Finally, the increased focus on guidance for specific measurement procedures and techniques will further harmonize the approaches and improve the reliability of results.

ICRU recognizes the role of ICRP in developing concepts and practical guidance for practical radiation protection. The role of both Commissions complements each other and, if beneficial for users of the recommendations, both Commissions collaborate as documented by the reports ICRU 57 and ICRP 74.

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