



*IRPA Regional Congress on Radiation  
Protection in Central Europe*

*Bratislava, Slovakia, September 22 - 26, 2003*

# **Proceedings**



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# IRPA REGIONAL CONGRESS ON RADIATION PROTECTION IN CENTRAL EUROPE



**Clearance levels and material release  
Environmental impact assessment of workplaces resp.  
facilities with radiation sources**

**Bratislava, Slovakia  
September 22 - 26, 2003**

*organized by*

**Slovak Society for Nuclear Medicine and Radiation Hygiene**

*in cooperation with*

Austrian Association for Radiation Protection  
Croatian Radiation Protection Association  
Czech Society for Radiation Protection  
German-Swiss Radiation Protection Association  
Health Physics Section of the Roland Eötvös Physical Society, Hungary  
Italian Radiation Protection Association  
Radiation Protection Section of the Polish Society of Medical Physics  
Romanian Society for Radiological Protection  
Radiation Protection Association of Slovenia

*under the auspices of*

Ministry of Health  
Ministry of International Affaires  
Ministry of Education  
Nuclear Regulatory Authority of the Slovak Republic  
Slovenske elektrarne, a.s.

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# Scientific programme

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### **General aspects of radiation protection, legislation**

- Future harmonisation of legislation, concepts and methods

### **Impact of workplaces with radiation sources on the environment**

- Environmental radiation protection - national legal basis, international recommendations
- Environmental radiation protection - concept and philosophy
- Material release and discharges from nuclear installations
- Discharges from non-nuclear workplaces with ionizing radiation sources
- Remediation

### **Health risks and influence of practices**

- Normal operational conditions, incidents, accidents, treatment
- Risk reduction resulting from nuclear installation reconstruction

### **Optimization of radiation protection**

- Quality assurance/quality control
- Safety culture
- Education and training
- Lessons learned from accidents

### **Biological effects of ionizing radiation**

### **Radiation protection in medicine**

### **Natural sources of radiation**

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IRPA Regional Congress on  
**Radiation Protection in Neighbouring Countries of Central Europe 2003**

Bratislava, Slovakia  
September 22 – 26, 2003-11-10

**Foreword**

Dear colleagues,

The Slovak Society of Nuclear Medicine and Radiation Hygiene, as the member of IRPA, organized, the next in turn, Regional IRPA Congress on Radiation Protection in Neighbouring Countries of Central Europe, in Bratislava in the days from 22 to 26 September 2003.

This international meeting was organized in cooperation with the Austrian, Croatian, Czech, German-Swiss, Hungarian, Polish, Romanian and Slovenian Associate Societies of IRPA. For the congress venue the excellent conditions of the Holiday Inn hotel were selected with good services and comfortable spaces. 145 registered participants from 16 countries attended the congress and they presented 67 oral and 54 poster presentations.

As with previous IRPA meetings the scientific content was broad, because the radiation protection is a dynamic profession and the consequences of research, as well as of the changing characteristics of radiation sources used and international standards developed, had to be presented.

The scientific work of the congress was divided into 8 sections:

General Aspects of Radiation Protection	19
Impact of Workplaces with Radiation Sources on the Environment	13
Environmental Monitoring	19
Dose assessment	18
Natural Sources of Radiation	21
Biological Dosimetry and Health Effects	13
Radiation Protection in Medicine	10
Optimisation of Radiation Protection and Quality Assurance	8

The numbers refer to the number of papers presented under the given topic.

During the congress different ways of material release, especially for nuclear installations decommissioning purpose were discussed, as well as the instrumentation used for giving the evidence that the release criteria has been met. Environmental impact resulting from clearance activities in Europe are kept at the very low level, even when the quantity of released material is rather high. Facing decommissioning of significant amount of nuclear power plants in next decade, good release practice becomes to be a very important topic.

The invited lecture on dosimetry issues for diagnostic radiology was followed with great interest. The main recommendations of the Code of practice for dosimetry in

diagnostic radiology were pointed out. This material is under final review and will be published by IAEA in 2004.

Several practical possibilities for dose reduction in mammography and some new statistical approaches were also presented. Three other invited lectures dealt with the implication of new ICRP Recommendations into the practice of radiation protection legislation, and emergency preparedness.

The significant role and results of European Radiation Dosimetry Group activities were explained and the harmonisation of procedures for internal dose assessment in EU discussed on the results of intercomparison exercises. The biological effects of ionising radiation in very low dose ranges reflected in the presented papers the latest development of the research.

The responsibilities of radiation protection specialists in activities concerning the education and communication with the public has been explained. International as well as national systems of education in the field of radiation protection were confronted with requirements for experts and. The lack of experts and their ageing was recognized in some countries. Actual problems connected with natural radiation and the possibilities of improving public awareness in radon issues were referred. Quality assurance was the important topic of several papers presented.

I hope, that the exchange of experiences at the international level as well as the official and also private discussions during the congress will help to all participants in their future work and will remain in their memories, together with the good atmosphere of the meeting.

The congress was supported by the product exhibitions that were regarded as an important integral part of the scientific programme. A variety of social events have been organized for scientific delegates and social participants. These included city tour in the old Bratislava, welcome party and congress dinner, and the visit of opera performance in the Slovak National Theatre.

All full presentations received until now are gathered in the Proceedings presented on CD-ROM and we hope that the reading will be convenient for all of you.

I would like to express my cordially thanks to all members of the Scientific and Organisation Committee, to the Sunway Agency and staff of the Holiday Inn for their excellent and immense amount of work that they have put in to make the meeting possible.

The representatives of Romania and Poland indicated their willingness to organise the next meeting in the year 2005. After discussion and because of the more convenient date of the Polish proposal, all societies agreed that the next IRPA Regional Congress would be organized by the Polish Society in Wroczlaw, from 14 to 19 of June 2005.

RNDr. Denisa Nikodemova, PhD.  
Chairman of congress

# EUROPEAN RADIATION DOSIMETRY GROUP: HISTORY, STATE OF ART, PERSPECTIVES

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## **History**

European Radiation Dosimetry Group (EURADOS) was established in 1981 with the original aim: “*to advance the scientific understanding and the technical development of the dosimetry of ionizing radiation in the fields of radiation protection, radiobiology, radiation therapy and medical diagnosis by the simulation of collaboration between European laboratories, specially those of the European Communities*”. EURADOS was formed to satisfy the widely perceived requirement to set-up cooperation on research in dosimetry. It was preceded by the creation of groups focused on drawing-up well established protocols and collecting basic data sets to improve traceability in neutron dosimetry radiation therapy and radiation biology research.

In the field of radiation protection, the activities of EURADOS were closely associated with the research programmes carried out under the auspices of the European Communities (EC) within the 2<sup>nd</sup> and 3<sup>rd</sup> Research, Technology and Development (RTD) Framework Programmes (FWPs). The increase of cooperation was achieved mainly by setting up working groups (WG) which addressed specific problems of ionizing radiation dosimetry requiring well defined tasks to be carried out including e.g. the dissemination of scientific and practical information, the organisation of intercomparisons and joint topical studies, the setting up of training activities, the improvement of measurement protocols, and quality assurance in numerical dosimetry.

WG operating in connection with EURADOS program are listed in the Table 1. To demonstrate the success of EURADOS in fulfilling its objectives it should be added that in the period up to 1996:

1. 8 International workshops was organised under EURADOS auspices.
2. 11 International intercomparisons were performed under EURADOS auspices in different actual topics of dosimetry.

3. More than 30 reports related to the results obtained within WG activities were published.
4. EURADOS contributed also to 7 training programmes in the frame of European Radiation Protection and Training (ERPET) program.

Table 1: Working groups of EURADOS established before 1995

No.	Topic	Years
1	Microdosimetry in and for radiation protection	82-89
2	Skin dosimetry	82-86
3	Applied TLD for routine individual monitoring	82-88
4	Numerical dosimetry	> 82
5	Application of track etch detectors to neutron dosimetry	85-89
6	Assessment of internal dose	82-96
7	Spectrometry of radiation in working environment	82-01
8	Individual dosimeters for penetrating radiation	90-94
9	Criticality accident dosimetry	89-94
10	Basic physical data for gas ionising dosimetry	91-95
11	Radiation exposure of air crews	92-96
12	Environmental monitoring	> 94

Key milestones of EURADOS activities since the beginning up to now are presented in the Table 2. Since 1996, the start of the 4<sup>th</sup> RTD FWP, EURADOS activities have been mainly confined to two working groups within concerted actions established separately with the EC (Numerical dosimetry, radiation spectrometry in mixed neutron-photon fields and throughout a joint concerted actions carried out with the other two scientific organisations EULEP (European Late Effects Project) group and UIR (Union Internationale EURADOS de Radioécologie). This action is aimed at reaching “*an integrated approach to radiation protection covering dosimetry and biological effects*”. A first steps forwards this end was achieved by editing the joint “EULEP- EURADOS-UIR Newsletter” and 4 concerted actions on different common topics.

#### **General information. State of art.**

Referring to the current constitution, EURADOS operates through:

1. Voting members (one representative per laboratory) elected for four years. Now, there is 47 voting members from 23 countries, including newly countries of Central and Eastern Europe, which were elected during the GA 2003.
2. The General Assembly (GA) of voting members which has full executive power and which meets at least once every two years (see also Table 2).
3. A Council of ten to fifteen members elected by terms of three years by the GA, with full management and administration power taking into account GA recommendations. After the GA 2003 there is 12 members of the council, chairman (P. Pihet, IRSN), vice-

chairman (Ch. Wernli, PSI), secretary (H. Schuhmacher, PTB), and treasurer (H. Zoetelief, TUD) forms a standing rate committee of EURADOS.

Table 2: Key milestones of EURADOS activities since 1990

1981	Registration of EURADOS constitution
■ 1990	9th General Assembly in Lisbon
■ 1991	1st revision of EURADOS constitution
■ 1991	10th General Assembly in Dublin
❖ 1992	Start of the first project funded under 3rd FP 1992-1995 (EURATOM)
■ 1992	11th General Assembly in Paris
■ 1994	12th General Assembly in Strasbourg
❖ 1997	Start of the coordination of the EULEP-EURADOS-UIR JCA under 4th FP 1997-1999
■ 1997	1st publication of the joint EULEP-EURADOS Newsletter
1999	Annual Meeting in Braunschweig (creation of a Task Group to propose changes in the future operation of EURADOS)
■ 2000	Validation of the Task Group conclusions by the Council in Fontenay-aux-Roses
❖ 2001	Start of the project "Radiation Dosimetry Network" under 5th FP 2001-2003
■ 2001	13th General Assembly in Braunschweig (update of membership, election of a new Council, 2nd revision of EURADOS constitution)
■ 2001	Creation of the EURADOS Web Site
■ 2002	14th General Assembly in Braunschweig.
■ 2002	Submission of an Expression of Interest towards the 6th FP
2003	Start the approach of a self-sustaining network

4. Topical working groups set-up by the council with associated members (more than 70 at the moment) selected according to their scientific expertise. Currently active working groups are presented in Table 3. In addition, EURADOS is keeping close relationships with different projects and concerted actions. They are also presented in Table 3.

Table 3: Working groups (WG) and concerted action currently active within the framework of EURADOS

Action	Topic	Remarks
WG 1	Irradiation facilities and special equipment for dosimetry research	
WG 2	Harmonisation of individual monitoring in Europe and information on new techniques in the field	
WG 3	Collaboration in environmental radiation monitoring	
CA 1	Numerical dosimetry (QUADOS)	5 <sup>th</sup> FP
CA 2	Internal dosimetry – together with EULEP	5 <sup>th</sup> FP
CA 3	Aircraft crew dosimetry – supported by DG TREN EC	

5. Important current action of EURADOS is organisation of 9<sup>th</sup> Neutron Dosimetry Symposium (Delf, 28.9.-2.10.2003) and preparation of the 10<sup>th</sup> Symposium on the same topic.

## **Future actions**

Since 1996, the start of the 4<sup>th</sup> RTD FWP, EURADOS activities started to be coming weaker. The GA held at 1999 decided to prepare a discussion document which would investigate the changes necessary to restore full capacity of EURADOS. Such document was prepared the same year [1], it formulated recommendations for improving the EURADOS operation abilities and increasing the resources. This document became the base for long period of reflexions, exchange of ideas and consequential recommendations. All this period was also combined with the preparation of EURADOS position vis-à-vis 6<sup>th</sup> FWP.

EURADOS was looking to maintain the capacity of setting-up and maintaining permanent networks of experts, and reference and research laboratories in order that appropriate specialist groups can be organised to solve problems identified within EURADOS or on request of external bodies and, if necessary, identify research, which needs to be supported.

Discussion during the last GA and CM (see Table 2) and, also, with many external bodies crystallize the current position, perspectives and future activities of EURADOS in the following way:

1. Several key thematics were identified, they are enumerated as follows: Environmental radiation monitoring, External/internal dosimetry, Protection of workplace, Aircraft crew dosimetry, Dosimetry fundamentals, Dosimetry for radiobiology, Medical radiation physics.
2. EURADOS will further support the existence and/or creation of WG in selected topics, deduced from those mentioned in item 1. It is expected that the basic support for them would be further related to general support coming to EURADOS from a DG.
3. Consultations with relevant stakeholders (DG, RTD, DG TREN, IAEA, ICRU, ICRP, ISO...) will be undertaken to specify more deep and clear the position of EURADOS in European Research Area in the field of radiation dosimetry.
4. EURADOS became, after 2003 enlargement, unique platform for further specification analysis and discussion within the cluster groups investigating the feasibility to prepare relevant projects and propose them furthermore in the frame of 6<sup>th</sup> FWP.

## **REFERENCES**

1. EURADOS. Proposal of changes in the future operation of EURADOS. Task group report, July 1999.



# **IRPA Regional Congress on Radiation Protection in Central Europe**

**Bratislava, Slovakia, September 22-26, 2003**

*INIS (International Nuclear Information System)*

*subject coverage of the Congress topics*

**Zsolt Stanik**

*IAEA, Wagramer Str. 5, P.O.Box 100, A-1400 Vienna, Austria*

**Environmental Aspects of Siting of Nuclear Installations** *(includes fission reactors, fission fuel cycle facilities and all other nuclear installations and facilities)*

- selection criteria, suitability studies and environmental impact theoretical studies under normal operating conditions

**Environmental Aspects of Radioactive Releases from Nuclear Installations** *(includes fission reactors, fission fuel cycle facilities and all other nuclear installations and facilities)*

- environmental implications for ecosystems resulting from generation, on-site treatment and release of radioactive substances from nuclear installations
- monitoring and transport of radioisotopes in soils
- monitoring and transport of radioisotopes in surface waters
- monitoring and transport of radioisotopes in earth's atmosphere
- personnel dosimetry and medical surveillance

*Note: covers are also environmental aspects of chemical and thermal releases from nuclear installations*

**Environmental Aspects of Design and Hypothetical Accidents at Nuclear Installations** *(includes fission reactors, fission fuel cycle facilities and all other nuclear installations and facilities)*

- environmental consequences predicted from the analysis of design basis or hypothetical accidents and performance of safety systems for nuclear installations including those involving handling and transport of radioactive materials
- environmental consequences of real accidents at nuclear installations

**Radiation Protection Procedures**

- procedures designed wholly or primarily to provide radiation protection for man
- prevention of contamination or procedures for decontamination, including chemical decontamination of materials, structures and equipment
- personnel monitoring and radiation monitoring (e.g., in nuclear facilities, industry, radiotherapy, x-ray diagnostics, nuclear medicine) for both patients and medical personnel

- medical surveillance of personnel exposed to ionising radiations in conformance with national or international radiation protection regulations or recommendations
- population dose estimates, collective dose and dose commitments as a result of nuclear accidents or from contaminated food
- calculation and measurement of absorbed doses in man, animals, plants and other biological systems at all levels, as well as in tissue-equivalent materials and phantoms
- radiation protection standards
- emergency planning
- radiation measuring instruments

## **Biological Effects of Ionizing Radiation**

- **Effects of External Irradiation on Biochemicals, Cell and Tissue Cultures, Microorganisms, Plants, Animals and Man**
  - effects of ionizing radiations (including immunological consequences, acute, and late effects) on man
  - relative effects of irradiation procedures, doses, dose rates, Relative Biological Effectiveness (RBE), Linear Energy Transfer (LET) and quality factors
  - modification of effects of such radiations due to various response modifying factors, such as radioprotective or effect-enhancing substances or irradiation conditions; side effects (e.g. toxicity) of such substances
  - side and late effects of such radiations in medical diagnosis and therapy
  - epidemiological studies of possible radiation-caused illness
- **Effects of Internal Irradiation, Radioisotope Kinetics and Toxicity in Microorganisms, Plants, Animals and Man**
  - acute and late effects of absorbed or incorporated radioactive materials
  - internal source evaluation
  - side and late effects, including toxicity, of the use of radioisotopes in bound or unbound form in diagnosis and therapy
  - radioisotope kinetics, localization, uptake and elimination of radioisotopes at all levels (subcellular, tissue, organ and whole organism)
  - contamination and decontamination (both internal and external)
  - use of chelating agents or complex forming agents, modifying factors and radioprotective substances, e.g. EDTA, DTPA, stable iodine
  - epidemiological studies of possible radioisotope-caused illness

## **Legal Aspects**

- **Radioactive Materials and Radiation Sources**
  - legal aspects, including licensing and inspection of prospecting, production, handling, operation, trade, transfer and supply of radioactive materials and radiation sources

- **Nuclear Installations**
  - legal aspects, including licensing and inspection, of siting, construction, operation and decommissioning of nuclear installations
  - legal aspects of trade, transfer and supply of nuclear installations and equipment
  - legal aspects of radioactive effluents from nuclear installations
  - legal aspect of emergency planning
- **Radiation Health**
  - legal aspects of protecting personnel and members of the public
  - legal aspects of protecting the environment against contamination from the operation of nuclear facilities
  - legal aspects of direct or indirect applications of radioisotopes and radiations to man (e.g. medical and industrial applications, food irradiation, radiation from consumer products)
  - legal aspects of emergency planning
- **Management, Transport and Storage of Radioactive Materials and Waste**
  - legal aspects of national or international transport of radioactive materials and wastes by any means, and accident prevention
  - legal aspects of waste treatment
  - legal aspects, including licensing and inspection, of storage of radioactive materials, and of temporary or ultimate storage of radioactive wastes

## **INIS Contact Addresses**

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# **INIS**

## **International Nuclear Information System**

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Subject Control Unit

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**TI**

**(Title):** 5.4. Human population dose from artificial radiation sources in environment. 5.4. Davkova zatez lidske populace umelymi zdroji zarení v zivotnim prostredi

**AU**

**(Author):** Cejnar, F

**TYPE**

**(Type of Document):** Book

**SO**

**(Source):** Kovar, Z. Spurny, F. Spurny, Z. Novotny, J. Cejnar, F. Advances in ionizing radiation dosimetry. Pokroky dozimetrie Ionizujiciho zarení. Prague (Czechoslovakia). Academia. 1984. 340 p. p. 202-211. 123 figs 12 pages of photographs

**AB**

**(Abstract):** Radiation doses of the human population from the operation of nuclear power plants, from fuel reprocessing and other artificial radiation sources are evaluated. It is shown that more than 50% of doses from nuclear power generation is given by the global dispersion of long-lived radionuclides  $^{14}\text{C}$ ,  $^{85}\text{Kr}$  and  $^3\text{H}$ . A comparison of natural and artificial radiation sources shows that the greatest proportion of radiation doses from artificial sources comes from the use of ionizing radiation in medicine. (Ha)

**DEI**

**(Indexer-Assigned Descriptors):** body burden; fission products; global aspects; human populations; nuclear facilities; nuclear medicine; radiation doses; radioactive effluents; radioisotopes; reprocessing;

**DEC**

**(Computer-Assigned Descriptors):** isotopes; materials; medicine; populations; radioactive materials; radioactive wastes; separation processes; wastes;

**YEAR**

**(Publication Year):** 1984

**LA**

**(Language):** Czech

**C1**

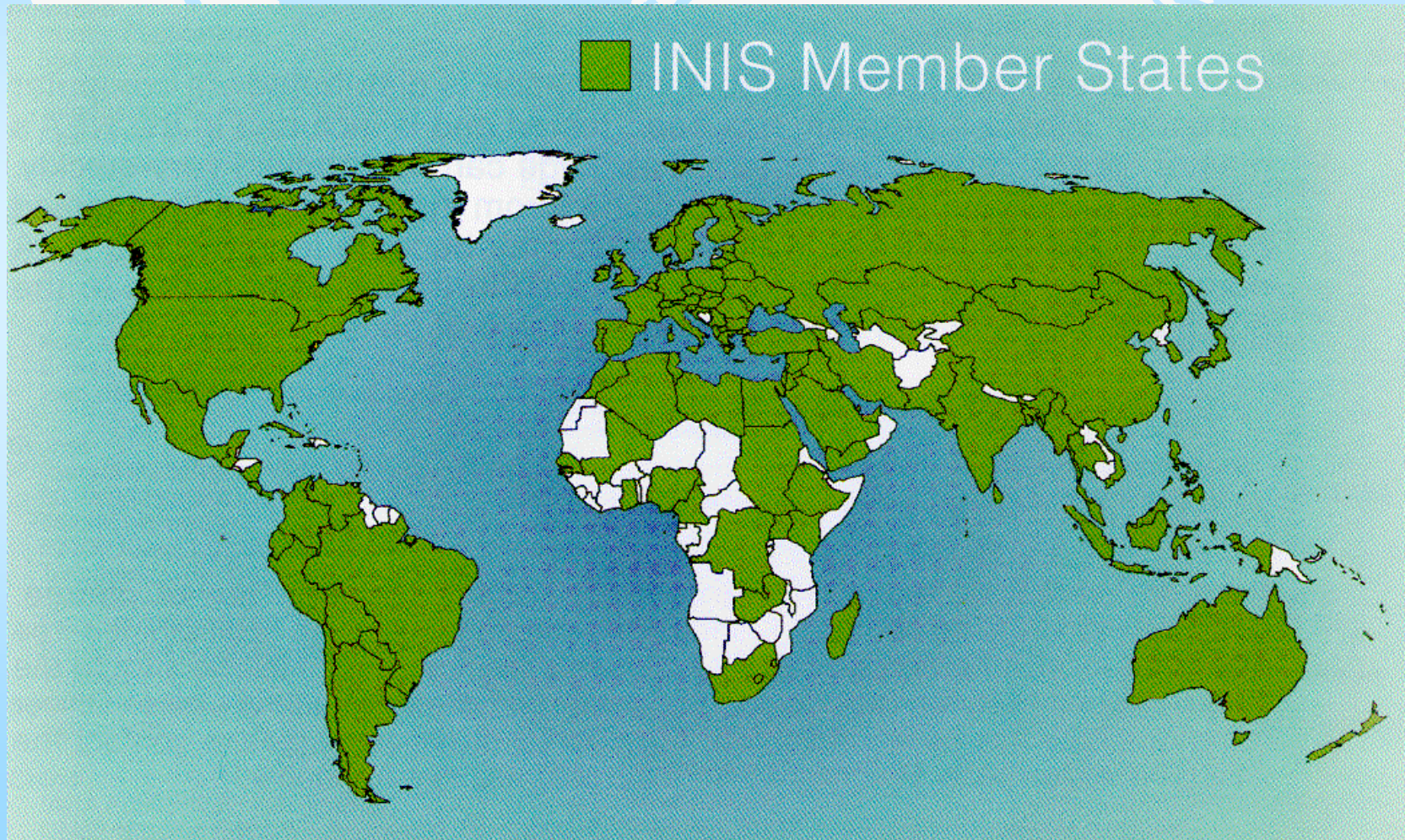
**(Primary Category Code):** C5500

**CD**

**(Explanations of Category Codes):** Personnel Dosimetry and Monitoring

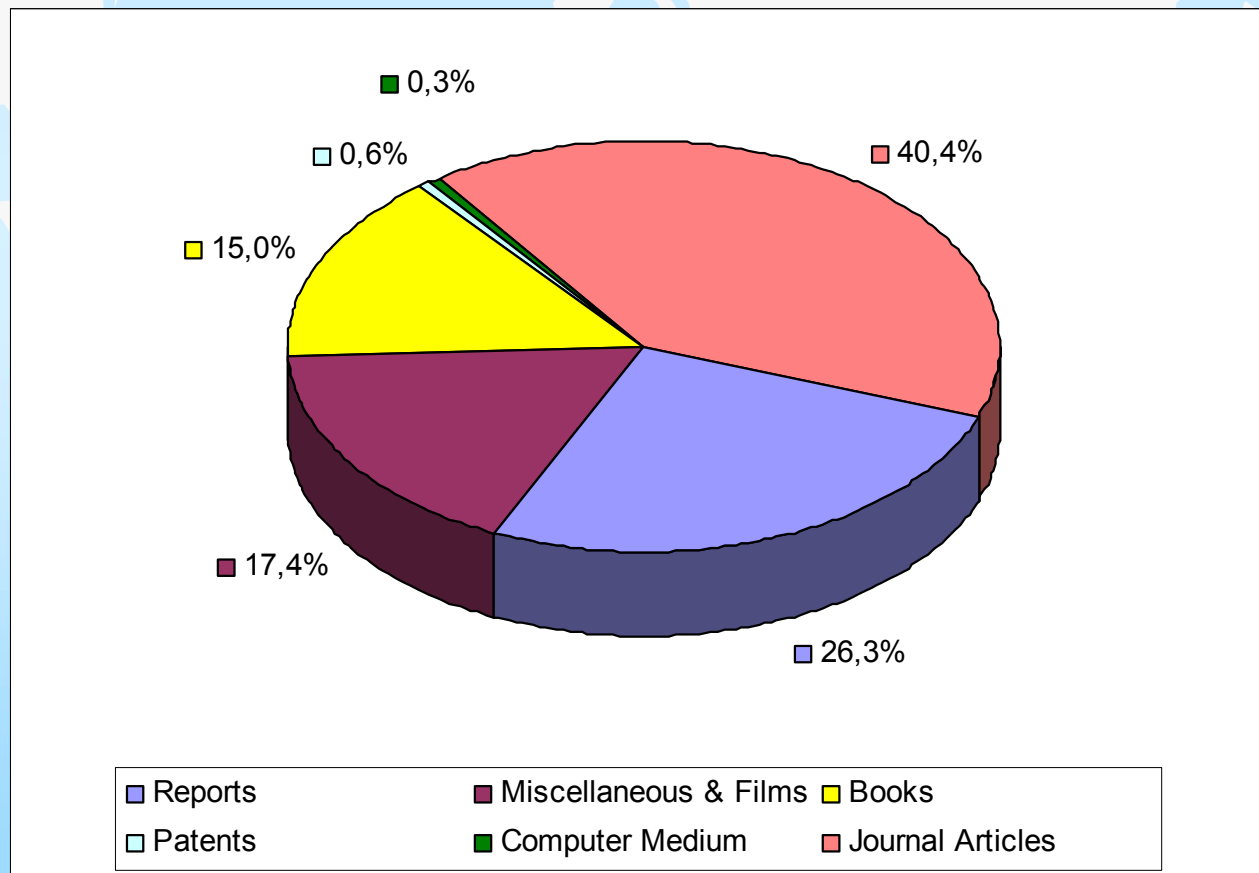


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- ◆ **Environmental Aspects**
- ◆ **Radiation Protection**
- ◆ **Biological Effects of Radiation**
- ◆ **Legal Aspects**

# **Environmental Aspects**

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- ◆ **Siting of Nuclear Installations**
- ◆ **Radioactive Releases from Nuclear Installations**
- ◆ **Design and Hypothetical Accidents at Nuclear Installations**



# **Radiation Protection (RP)**

- ◆ **Procedures to provide RP for man**
- ◆ **Prevention of contamination**
- ◆ **Procedures for decontamination**
- ◆ **Personnel and radiation monitoring**
- ◆ **Dose calculation and measurement**
- ◆ **Radiation protection standards**
- ◆ **Emergency planning**

# Biological Effects of Radiation

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- ◆ **Effect of external irradiation on biochemicals, cell and tissue cultures, microorganisms, plants, animals and man**
- ◆ **Effects on internal irradiation, radioisotope kinetics and toxicity in microorganisms, plants, animals and man**

# Legal Aspects



- ◆ **Radioactive materials and radiation sources**
- ◆ **Nuclear installations**
- ◆ **Radiation health**
- ◆ **Management, transport and storage of radioactive materials and waste**

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**ACCESSION OF THE SLOVAK REPUBLIC TO EU  
– NEW LEGISLATION, NEW CONDITIONS AND NEW  
APPROACHES IN THE AREA OF EMERGENCY  
PREPAREDNESS**

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**1. Introduction**

The safety of nuclear facilities and sources of ionising radiation, radiation protection, the safe management of radioactive waste and the safe transport of radioactive materials are of great importance to individuals and society and to the environment in the state of use and in other states. The emergency preparedness is one of important issues related to the nuclear safety particularly in case of nuclear or radiation incident or accident. Since its establishment in 1993 the Nuclear Regulatory Authority of the Slovak Republic (UJD) has been making efforts to follow international practices in the area of emergency preparedness and planning. Starting its regulatory activities UJD followed former Czechoslovak legislation which were transformed step by step to comply with international practices. At present new legislation regulatory tools are developed, implemented and used to supervise and control peaceful use of atomic energy in the country. Efforts have been made to align this new legislation and practises with EU directives and IAEA recommendations.

**2. Legislation in Emergency Preparedness**

At the beginning former Czechoslovak Atomic Act and Decrees of Czechoslovak Atomic Energy Commission were used to regulate emergency planning in nuclear power area. There were a few paragraphs only concerning this area, which did not specify and cover clearly enough all necessary aspects of emergency planning. On- and off-site emergency plans were not unambiguously specified and they did not treat emergency transport orders at all.

Therefore regulations of US NRC [1] and recommendations of IAEA [2,3] were used as a guidance to prepare emergency plans of nuclear facilities by the operators.

The national emergency management was represented by the National Emergency Commission for Radiation Accidents (NECRA) established by decision of Slovak Government. This commission, however, did not have any technical or professional support for decision making in case of emergency. Therefore UJD, based on experience of countries with developed nuclear power, built its emergency response centre (ERC) to be able to cope with nuclear incidents and accidents and provide a technical support for decision making on a national level. This ERC has been running since 1995.

Meanwhile intensive works on preparation of new Slovak legislation concerning nuclear power were in course. As a result a new law on peaceful use of atomic energy (Atomic Act) [4] was approved and entered in force in 1998. Events at nuclear facilities and during transport of nuclear materials were specified and emergency planning was broadly treated in relevant paragraphs of this new law. Consequently a decree of UJD on emergency planning for the case of incident or accident was issued in 1999 [5]. It is to be said that during the preparation of this decree naturally recommendations of IAEA and directives of EU concerning emergency planning [e.g. 6,7,8] were taken into account. This decree specifies in detail the contents, i.e. the way of preparation, organisation arrangements, event classification, warning and notification procedures, frequency of exercising of on site and off-site emergency plans as well as of emergency transport orders, defines emergency planning zones and set up duty to inform public including the use of INES scale. Besides four phases of the accident- threat period, early phase, temporary phase and late phase - and corresponding basic countermeasures were defined and introduced in the part dealing with off-site emergency planning and basic intervention levels were published in the decree enclosure.

Not to forget the emergency response centre of UJD was fixed in the Atomic Act and duty for operators and authorities to provide data necessary to assess technological status of impacted facility and radiological situation was set down. UJD was appointed to be a contact point – national competent authority – for communication with IAEA.

Based on these new regulations all on- and off-site emergency plans of nuclear facilities in Slovakia were reviewed, remade and submitted for re-approval in 2000. For first time also emergency transport orders were prepared by companies dealing with transport of nuclear fresh and spent fuel as well as radioactive wastes. At present all these emergency plans are unified as to the form, structure and contents. The emergency plans are regularly

exercised starting from small-scale exercises on the level of NPP shift up to the large scale exercises when substantial elements of emergency organization structure are involved in. UJD takes part in these exercises either in position of inspector or observer or co-operates by means of its ERC as an active member of the relevant exercise.

### **3. Emergency arrangements**

Up to now the national emergency management consisting of three levels – nuclear installation –local level –national level - for the case of nuclear or radiation accident headed by NECRA was rather autonomous one. Due to the fact that Slovak Republic is going to access EU in 2004 all legal documents have to comply with EU directives and legal documents. Therefore last two or three years several missions coming from EU, NATO, made reviews and audits concerning emergency preparedness and planning in Slovakia in general. Based on results of these audits following new four important laws were issued in 2002:

- Act No. 129/2002 on Integrated rescue system
- Act No. 261/2002 on Prevention of serious industry accidents
- Act No. 414/2002 on Economic mobilization
- Act No. 387/2002 on State crisis management during peaceful and during war time

In the light of these new legal documents also the national structure of crisis management and emergency preparedness has been changed. Details and structure of emergency preparedness in nuclear area were untouched by these new laws and remained as the same as before as to the preparation of emergency plans, accident phases, and decision making levels, however, to comply with these new laws it was necessary to change the structure of emergency and crisis management in Slovakia as a whole. A new organisational hierarchy has been established. In frame of this new organisation of crisis management the nuclear emergency management has a role of expert body to support national emergency headquarters, which are further subordinated to the national security council being now the most important state organisation in decision making issues for any emergency and/or crisis situations in the country during both peaceful and war times.

### **4. Conclusion**

At present after five years of UJD experience of using new Slovak regulations and before entering to EU, which is foreseen since May 2004 a comprehensive review of Atomic Act and all relevant decrees is being made with the objective to fit them to EU directives. On

line with that also necessary organizational and technical arrangements in the area of regulatory activities including emergency preparedness are prepared. UJD participate actively in EU and IAEA projects to achieve the harmonization of its practices with EU and other countries with developed nuclear power and makes efforts to keep on a high level of nuclear safety to be a valuable member of EU.

## **5. References**

- [1] Criteria for preparation and evaluation of radiation emergency response plans and preparedness in support of nuclear power plants, NUREG-654, US NRC, 1980
- [2] IAEA safety series 50-sg-06: Operator preparedness for emergency situations at nuclear installations
- [3] IAEA safety series 50-sg-66: Preparedness of public administration bodies for emergency situation at nuclear installations
- [4] Act No. 130/1998 on peaceful use of nuclear energy,
- [5] UJD Decree No. 245/1999 on emergency planning for the case of incidents and accidents
- [6] 82/501/Euratom: Council Directive of 24 June 1982 concerning risks associated with severe accidents in certain industrial activities
- [7] 87/600/Euratom: Council Decision of 14 December 1987 on the setting up of a system of Community measures for rapid exchange of information in radiological emergencies,
- [8] 89/618/Euratom: Council Directive of 27 November 1989 on information provided to general public on health protection measures that need to be applied and on steps to be undertaken upon radiological emergencies.



## **SOME COMMENTS CONCERNING R&D AND EDUCATION IN RADIATION PROTECTION IN SLOVAKIA**

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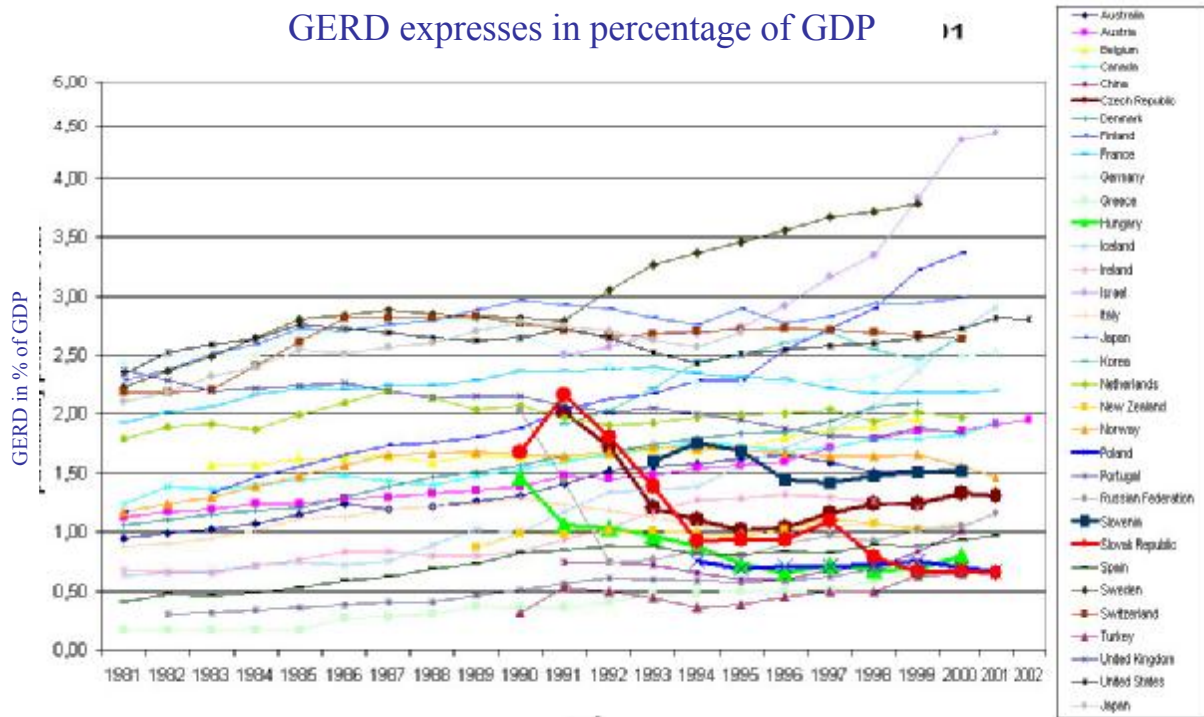
The ability to create, distribute and exploit knowledge is increasingly central to competitive advantage, wealth creation and better standards of living. According the EU documents, research and development (R&D) contribute about 25 – 50% to rapid economic growth. Indicators that capture the relationship between science, innovation and economic performance are crucial so that policy makers may make informed decisions, set priorities and address the challenges of the knowledge-based economy.

One of the most important indicators is Gross Expenditure on Research and Development (GERD) expresses in percentage of GDP. Fig. 1 depicts the evolution of GERD during the last two decades for OECD countries (1). The statistics for the transformed economics (Slovakia, Hungary, Poland, Czech Republic, and Slovenia) are available since 1990. From the graph one can see slow but systematic increase of the GERD in developed countries. The Nordic countries notably Finland and Sweden, appear to have the highest expenditures for R&D, they are leaders in the transition to knowledge-based economy (high investment in knowledge, rapid innovation). The US has the GERD constantly more than 2%. Below average growth in R&D expenditure in the EU is mainly due to slow and declining growth in the major European economies. Compared to OECD average growth (2.8%) over the 1991 – 99 periods, R&D expenditure grew by half or less in Germany (1.4%) and the UK (1 %). Among the OECD countries R&D expenditure declined only in the Slovak Republic, Hungary and Italy. In Slovakia the R&D funding is decreasing, uncertain and insufficient. GERD is the lowest from all compared countries and is has decreasing trend. It is alarming and the intention expressed in “Guidelines for the State’s science policy” to reach the GERD 3% in 2010 seems to be highly unrealistic.

Fig.1. Gross Expenditure on Research and Development (GERD) expresses in percentage of GDP during the period 1981 – 2001

## GERD expresses in percentage of GDP

11



This decreasing trend of R&D expenditure in Slovakia will reduce the scope for economic and social development. The number of university level educated and S&T highly skilled workers rapidly declines and young talented people are leaving country, looking for better opportunities. This decrease in Human Resources in Science and Technology (HRST - which grew significantly in EU countries, about 3 % annually) in Slovakia could definitely give rise the question if Slovak Republic belongs to European Research Area.

This situation is widespread through all research fields and education. The Society of Nuclear Medicine and Radiation Protection is aware of these alarming facts affecting basic research and education in radiation protection (RP) at all levels which in turn affects the ability to deliver safe and effective use of technologies involving radiation, such as diagnostic radiography, nuclear diagnosis and cancer radiotherapy. The funding of research and education drawn from the State's budget is insufficient to cover the requests for EU level. The result of this is practically no basic research in radiation biophysics, cellular and molecular biology to study the interaction between radiation and the DNA, cells, organs and the body, almost no epidemiological studies for quantification of risk associated with low and protracted exposures. Another consequence is the lack of qualified experts for educating the students and the doctors, dentists, medical physicists, technicians, nurses in radiation protection in medicine, in industry, the radiation protection officers in NPP, etc.

Slovakia has strong ambitions to be a member of the EU. Taking part in EU brings a lot of duties and responsibilities. In the field of RP the minimum request is to satisfy the criteria set out in the Basic Safety Standards, in 80/836/ Euroatom (protection of workers and the general population against dangers arising from IR), in last revision by Council Directive 96/29/EUROATOM, in Medical Exposure Directive (MED).

Currently, Slovakia is not able to satisfy these requirements. Today in Slovakia there is a need for TARGET PROGRAM devoted to these issues.

## **TARGET PROGRAM**

### **1. Research in the field of ionizing radiation**

It is really urgent to start the program in order to revitalize the basic and also the applied research on interaction between ionizing radiation and DNA, cells, organ., as well as the education programs in RP. Taking into account the facts that in Slovakia the funding of research and education drawn from the State's budget is insufficient, the only one realistic solution is in taking part in international research activities. There is a lot of opportunities, for instance in the EU Sixth Framework Program (6FP).

#### **EURATOM - Research and Training Program on Nuclear Energy (2002 – 2006)**

**International co-operation** represents an important dimension on the 6FP.

#### **RADIATION PROTECTION under 6FP**

##### **Quantification of risk associated with low and protracted exposures.**

Research is focused on epidemiological studies of suitable exposed populations, and on cellular and molecular biology research on the interaction between radiation and the DNA, cells, organs and the body.

##### **Medical exposures and natural sources of radiation**

To enhance the safety and efficacy of medical uses (radiodiagnosis, nuclear medicine, radiotherapy) of radiation and better understand, assess and manage natural sources of radiation.

##### **Protection of the workplace**

Improving the monitoring and management of occupational exposure in industries involving exposure to radiation

### **2. Education and training in RP**

The second part of TARGET PROGRAM has to be devoted to the systematic education in RP at different levels (high school, undergraduate, graduate, special training programs).

The ICRP 73 1996 (2) states: "one important need is to provide adequate resources for the education and training in RP for future professional and technical staff in medical practice.

The training program should include initial training for all incoming staff and regular updating and retraining”.

In Slovakia there is no University which teaches medical physics with the specialization for preparing the professionals in the field of radiotherapy, nuclear medicine, radiodiagnoses. On the other site the medical students during their study at the University get poor education and sometimes only very brief information about RP. If Slovakia wants to take part in the EU it is necessary to start to educate future professionals in the field of RP according the system developed by European Commission in the field of radiation protection, RP 116 (3).

## RADIATION PROTECTION 116

### I. Guidelines on education and training in RP for medical exposures

TRAINING AREA	DR MD	RT MD	NM MD	CD MD	DT	MD	RD	NU	ME
Atomic structure, production and interaction of radiation	M	H	H	L	L	L	M	L	M
Nuclear structure and radioactivity	M	H	H	L	-	-	M	L	M
Radiological quantities and units	M	H	H	M	L	L	M	L	M
Physical characteristics of the X-ray or therapy machines	M	H	L	M	L	M	M	L	H
Fundamentals of radiation detection	L	M	H	L	L	L	M	L	H
Fundamentals of radiobiology. Biological effects of radiation	M	H	H	M	L	M	M	L	L
Radiation protection. General principles	H	H	H	H	M	M	H	L	M
Operational radiological protection	H	H	H	H	M	M	H	M	M
Particular patient RP aspects	H	H	H	H	M	H	H	M	M
Particular staff RP aspects	H	H	H	H	M	H	H	M	M
Quality control and quality assurance	M	H	H	M	L	L	M	L	H
National and European regulations and standards	M	M	M	M	M	M	M	L	H
Suggested number of training hours	30 - 50	40 - 60	30 - 50	20 - 30	10- 15	15 - 20	40- 100	10- 15	40- 60

DR/MD = Diagnostic Radiology Specialists  
 RT/MD = Radiotherapy Specialists  
 NM/MD= Nuclear Medicine Specialists  
 CD/MD = Intervetional Cardiologists  
 DT = Dentists  
 MD = Medical doctors using X-ray

RD =Radiographers  
 NU =Nurses  
 ME = Maintenance Engineers  
 level of knowledge: L = low  
 M = medium  
 H = high

## **II. Recommendations regarding the course on RP in the curriculum of medical and dental schools.**

According to the EC MED, members States shall encourage the introduction of a **course on RP in the basic curriculum of medical and dental schools.**

This training should include all the basic RP knowledge needed by the prescriber who should be educated in the basic aspects of RP, specially justification of medical exposure and optimization (right choice of equipment|).

**In medical schools**, the main topics should be the **general aspects of patient protection** such as biological effects, justification of medical exposure, risk benefit analysis, typical doses per exam, etc., together with some basic knowledge of the advantages and disadvantages of the use of ionizing radiation in medicine. Medical students do not need specific training in the design and operation of the medical installations required for radiodiagnosis, NM and RT. This specific RP training will form part of their training program as residents to become specialists.

The case of **dental schools** is different. In addition to basic aspects the course of RP should also include all the specific training for the safe operation of X-ray systems for diagnostic purposes (principle of X-ray tube operation, radiographic imaging, film processing, quality assurance programs, occupational and patient dose control, etc.

A basic RP course should be also introduced **in nursing and paediatry schools**.

The duration of this RP training should be between 20 and 40 hours assuming a prior knowledge of radiation physics. 20 – 30 % should consist of practical or seminar sessions analyzing typical cases presented in clinical practice.

### **Conclusion**

Experience shows that when a country neglects it own R&D spending is at the risk of slowing its economic development. In Slovakia, radiation protection training and research have been neglected in recent years. In some important respects, co-operation through EU based research and training programs provides an opportunity to improve the situation.

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# SOME EXPERIENCE WITH THE RECENT DEVELOPMENT OF STANDARDS IN RADIATION PROTECTION

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## Abstract

*Conceptual issues in Radiation protection are today subject of a development being faster and more complex than some years ago. Scientific progress has to be incorporated into standards and legislative issues, but the time schedule is becoming tighter than before. This is because developments take place by different bodies, under different constraints and also in different administrative levels and dimensions. This lead to a situation that additional interactions takes place and issues of practicability have to be taken into account, disregarding irrational political issues. Some major issues in 1990 recommendation are not yet implemented and not properly used. Another issue is that standards for different exposures are not discriminating between different potential of dose reduction, but execute all in the same manner. As the discrepancy between conceptual and practical issues becomes more diverging than before, some more effort is needed to develop links between different types of standards (as recommendations and technical standards). In the present paper, the recent development for some modes of exposure is discussed considering issues as interaction of different types of standards, use of different dose quantities, hierarchy of limits.*

## 1. General

Radiation Protection Standards are developed by different organisations. As the objective of the different bodies is different, the results are not necessary the same. This lead eventually to some potential of misinterpretations and difficulties in application. Standards have the objective to prove uniform conditions, but the term “uniformity” is considered different in different standards, and this in turn might lead to difficulties.

As the names of the committees imply, the area covered by the different organisations is different. This in turn lead to the different approaches in objectives and hence in terminology, and requirement in accuracy. Meetings try to make-believe harmonisation (as Florence 02), but the process of harmonization is not fully developed. Therefore, the following approaches have to be brought in line for rules in radiation protection (see figure 1 next page)

## 2. Development of Standards

### 2.1 Scientific basis:

Data on deterministic effects are well proved /IC 91/, data on stochastic radiation effects include still some uncertainties / Pi 01, Sa 01, Ca 01/. These informations provide the biological basis of radiation protection, and the data from the original investigations are further evaluated by bodies as UNSCEAR /UN 00, UN 01/. The original data lead numbers in risk factors, but indicate also a substantial uncertainty ( see e.g. <http://www.rerf.jp/> )



## 2.3 Standards of International Organizations

The Basic Safety Standards of International Organizations /BS 96/ are based upon ICRP and ICRU, are focused only to radiation protection issues in depth, but are general in formal issues and apply not directly in some countries as EU members.

## 2.4 Legal approach

The Standards of the European Commission [EU 96a, EU 97] are also dealing with radiation protection, but have to be in line with formal general requirements. Translation in numerous languages lead to additional problems, and some terms are not very practicable<sup>1</sup>. National legislation has to take into account additional constraints, where e.g. they tend to be very “cautious” to lower certain figures in general.

## 2.5 Technical approach

The following national and international organisations issue technical standards in radiation protection/Gr 02/:

- a) European Committee for Standardization: CEN ([www.cenorm.be](http://www.cenorm.be))  
CEN/TC 114 Safety of machinery  
Safety of machinery - Ionising radiation
- b) European Committee for Electrotechnical Standardization:  
CENELEC ([www.cenelec.be](http://www.cenelec.be)) mainly medical field, X-ray
- c) International Organization for Standardization: ISO ([www.iso.ch](http://www.iso.ch))  
ISO/TC 85 Nuclear Energy (SC2 Radiation Protection).  
ISO has more than 200 Technical Committees, their scope ranging from TC 127: Earth moving machinery to TC 191: Animal (mammal) traps
- d) International Electrotechnical Committee: IEC ([www.iec.ch](http://www.iec.ch))  
IEC/TC 45 Nuclear instrumentation (SC45B Radiation Protection Instrumentation)  
IEC/TC 62 Electrical equipment in medical practice
- e) On a national level, organisations as DIN and ÖNORM are producing standards in close relation to international standards
- f) In addition, standards are also available for transport of radioactive material within the framework of “Safe Transport of dangerous goods” issued by international organisations, also applying different constraints and terminology

## 3. Examples

Some examples are presented below to show the relation between different standards.

### a. Language

Radiation protection standards have the task to convert the findings gained in scientific investigations (including cautions conclusions, expressing uncertainty of finding, requiring further investigations, etc) into clearly defined and executable rules, which in turn can be set

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<sup>1</sup> e.g. the definition of „Ionizing radiation“. A more appropriate definition can be found in the International BSS/IA 96/



into force by legislative means and which are executable by people unaware of the scientific background.

The phrases are therefore different, the examples below perhaps a little overdone to show the conversion of a scientific statement to a paragraph in a legal document:

*Scientific:* The information available at present suggests that there is some evidence that a dose of less than 1 mGy might not lead to the detectable deleterious effects.

*Recommendation:* It might be advisable to keep the dose in general below 1 mGy.

*Legal:* The dose must not exceed 1 mGy.

Interpretation is therefore occasionally difficult, and the accuracy of the number is much lower than it looks.

## b. Quantities and Units

There are different kinds of quantities, which are defined in order to fulfil specific requirements. In particular, the “new quantities” created many years ago are still not yet implemented. A few examples related to the dose are shown in brief in Table 1:

	Example	Unit	Properties
Basic Quantities (ICRU)	Absorbed dose D	J/kg, Gray [Gy]	generally applicable physical quantity not measurable directly
Limiting quantity (ICRP)	Effective dose E	Sievert[Sv]	for stochastic effects only not measurable directly
Operational quantities (ICRP)	Ambient equivalent dose H*(10)	Sievert[Sv]	Conservative approach of E Measuring quantity
Quality factor (ICRU)	Q	1	based on LET
Radiation weighting factor (ICRP)	w <sub>T</sub>	1	based on biological effects
Special quantities	Dose area product	[Gy.m <sup>2</sup> ]	medical radiography
	Committed dose		Integrated dose

It has to be stated that international recommendations and standards are not excessively clear in these issues.

Some confusion can be demonstrated by the use of different kinds of quantities in limitation of dose:

Based upon ICRP, the BSS and the EU-guideline define for members of the public:

The estimated average doses to the relevant critical group of members of the public shall not exceed the following limit of 1 mSv/a, effective dose. This implies that the dose clearly refers to persons.

In technical Standards, however (DIN 6814,) a term “Ortsdosis” (dose at a certain site) is coined and defined as: Equivalent dose, *measured* at a certain site (DIN 6814)".

This quantity is adopted in ÖN S 5212 /ÖN 02/ and used as limiting quantity to prove compliance with limits. The Austrian Radiation Protection ordinance will probably also follow the procedure. The development is adopted from the period where no distinction between operational and limiting quantities was made. Also used since

many years, the quantity used is no primary limit (annual dose) but a derived limit (dose per week)

The development is not in line with /EU 96/, because the effective dose is used correctly as limiting quantity, comprising of external exposure, and internal exposure by inhalation and ingestion. Technical standards, however, use an operational quantity as limiting quantity. This leads to the situation that

- a figure to be measured ( by definition retrospective) is used for design of protective measures of an installation
- a dose of a individual is changed to a dose at a certain site, which in turn is biasing optimisation (see 2.3 below)

Technical standards do therefore apply quantities in another sense then initially intended.

### **c) Biasing Optimization**

Primary standards require, as one of the key issues, optimisation of protection. This requirement is one of the most misunderstood issues and the procedure is often not straightforward. Optimisation is therefore difficult to execute as authorities are usually in charge of radiation protection only and not responsible for general safety and hence for possible non-radiological hazards. For the same reason, it happens that technical standards incorporate an additional bias by straightforward procedures leading to over-conservative approaches.

An example of such procedure is shown below for the limitation of the dose of a member of the population by medical X-ray equipment.

As a limiting quantity, the dose at site under consideration is taken instead of dose of an individual (as shown in 2.2)

In design, parameters are set by standards leading to bias of optimisation by numerous “cautious” assumptions

a) design assumptions:

Design high voltage based upon equipment characteristics rather than on medical requirements (e.g. a tube potential of 150 kV have to be taken in any case even when investigations have to be done with 85 kV. The use of then potential of 150 kV would lead to useless images and is in contradiction to other EU guidelines as /EU 96b/ operational parameters are checked without taking into account their sensitivity (e.g. a increase of the workload for a factor of two requires an additional layer of 0,1 mm lead for the primary beam at 150 kV, but a change of the tube potential from 75 kV to 100 kV requires an additional layer of 0,7 mm lead. Both layers are required for the same shielding effect.

distribution of scattered radiation assumed as homogenous, maximum scattering factor chosen over-conservative for a factor of two

anode angle (and hence spectrum) not taken into account

b) testing and operational assumptions

equipment to be operated at maximum high voltage (never used in normal operation)  
required accuracy do not take into account of sensitive and nonsensitive parameters

small field size taken as to be representative

Due to design constraints mentioned above, the dose at the site of interest will never numerically identical with the design figure. But it is still assumed that a single person remains their full lifetime on the considered point, e.g. at the outer wall of the ordination.

#### **d) Dependence of acceptable uncertainty of assessment of a number in relation to a limit to the used quantity**

International standards do not consider the question to prove compliance with limits, which can be done by modelling (prospective approach) and by measurement (retrospective approach). Both approaches include some bias (see above) but include also some uncertainties. However, the uncertainty of the limit is zero by definition, even when the number is expressed in one representative digit (e.g. 1 mSv/a), but the number must not exceeded (see 2.1). When the compliance with the limit has to be proved by measurements of dose or activity, the result includes both a number and an uncertainty. e.g.  $A \pm \Delta A(1\sigma)$ . Some work was done, which can be found in Austrian Standards ÖN 5250 and ÖN5255 /ÖN 02/ how the total uncertainty can be set in Relation with the limit. The hierarchy of the limits lead to different acceptable requirements in uncertainty / see Ts 02 and 2.5/

#### **e) Hierarchy of Limits**

There is a clear hierarchy of limits, as well described in /IA 86/ where the following distinction is made:

- Primary limits
- Secondary limits
- Derived limits
- Authorized limits
- Operational limits

This hierarchy seems of formal nature only and has apparently no consequences in most cases. However, one application is important where this hierarchy not actually taken into account. This is when compliance with the limit is to be proved by measurements ( see /2.4/). As only the primary limit is a value not to be exceeded, the question of the relation of the result and his uncertainty with the limit has to be discussed. In all other cases, the (lower than primary) limit is related with the primary limit by a model, e.g.

$$\text{dose } [\mu\text{Sv}] = \text{dose rate}[\mu\text{Sv/h}] \cdot \text{time } [\text{h}]$$

If this model is applied to prove the dose limit (e.g. the annual dose, being a primary limit) by accurate assessment of the derived limit: dose rate [ $\mu\text{Sv/h}$ ], or dose per week, the uncertainty of the result is governed, following the well known law of error propagation, by the larger uncertainty of the components of the product. Improvement of the uncertainty of one factor (the dose rate) will have very little influence to the total uncertainty when the uncertainty of the predicted time [h] is not known, and may range from 1 hour or 8766 h per year. To adopt a residence time of more then 8766 h per year is over- conservative, but even to take a full year is biasing optimisation. These facts imply that improvement of the uncertainty of the dose rate measurement will not improve the uncertainty of the annual dose at all.

## f) Potential of dose reduction

Recently, standards included additional pathways, but they do not distinguish in procedures, assessment and required accuracy between the associated potential of dose reduction. The following table shows the main characteristics:

Source	Dose	dose range	potential of dose reduction
Radon	High	Large	High
cosmic radiation in airplanes	Medium	Small	Low
terrestrial external exposure	Low	Low	No
medical	High	large	*)

+) new techniques (CT, IR) are associated with a high dose, but justified by high diagnostic and therapeutic benefits

## 4. Conclusions

The development in different standards is occasionally not consistent with the conceptual basis of radiological protection:

- 1) For administrative reasons, primary quantities are replaced by secondary or derived quantities because of easier assessment and monitoring. The compliance with primary quantities is often replaced by compliance with derived quantities
- 2) Requirement for the accuracy of the assessment of other than primary quantities (dose per week, activity concentration) are enhanced although contributing parameters (occupancy time, consumption) include large uncertainties by definition
- 3) Limitations are often expressed in terms of derived quantities just for easier assessment. This has no conceptual basis in the dose range of stochastic effects.
- 4) It is not distinguished between limiting and operational quantities
- 5) As the development of standards consumes too much time, a new generation of radiation protection standards is to be expected before the previous step is executed.
- 6) Harmonization of standards is a need in order to make progress in radiation protection practically executable in the future

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# EUROPEAN PROJECT FOR DEVELOPING GENERAL GUIDELINES FOR HARMONISING INTERNAL DOSE ASSESSMENT PROCEDURES (IDEAS)

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## Introduction

Several international intercomparison exercises on intake and internal dose assessments from monitoring data led to the conclusion that the results calculated by different participants varied significantly mainly because of the wide variety of methods and assumptions applied in the assessment procedure. Based on these experiences the need for harmonisation of the procedures has been formulated as an EU research project under the 5<sup>th</sup> Framework Programme (2001-2005), with the aim of developing general guidelines for standardising assessments of intakes and internal doses. In the IDEAS project eight institutions from seven European countries are participating using inputs also from internal dosimetry professionals from across Europe to ensure broad consensus in the outcome of the project.

## Work Programme

The project is divided into Work Packages (WP), one for each of the five major tasks. The structure of the project and the interaction between Work Packages are shown in Figure 1.

Work Package 1 entitled *Collection of incorporation cases* was devoted to the collection of data by means of bibliographic research (survey of the open literature), contacting and collecting data from specific organisations and using information from existing databases on incorporation cases. Two databases (the bibliographic database and the incorporation cases database) were to be prepared and some reference cases for the performance of Work Package 3 selected.

In Work Package 2 (*Preparation of evaluation software*) an existing computer code IMIE [1] was to be used as a platform for testing existing methods and approaches for bioassay data interpretation and methods developed in the project. This software was to be provided to the partners for the evaluation of reference cases. The current version of the computer code IMBA [2] was also made available to the participants to support the evaluation procedures. In this WP, different and new methods of data interpretation are studied and compared, the pilot program unit

of IMIE was developed and tested, and the procedures of input and output data from the incorporation cases database (WP1) implemented.

Work Package 3 deals with the *Evaluation of incorporation cases* by means of the software provided by WP2 and using the reference cases from WP1. Each selected case was to be evaluated by at least two partners. The evaluations are compiled in a database pointing out common assumptions for similar scenarios, applied models and parameters and procedures to assess uncertainties, handling outlying data and measurements below the limit of detection etc..

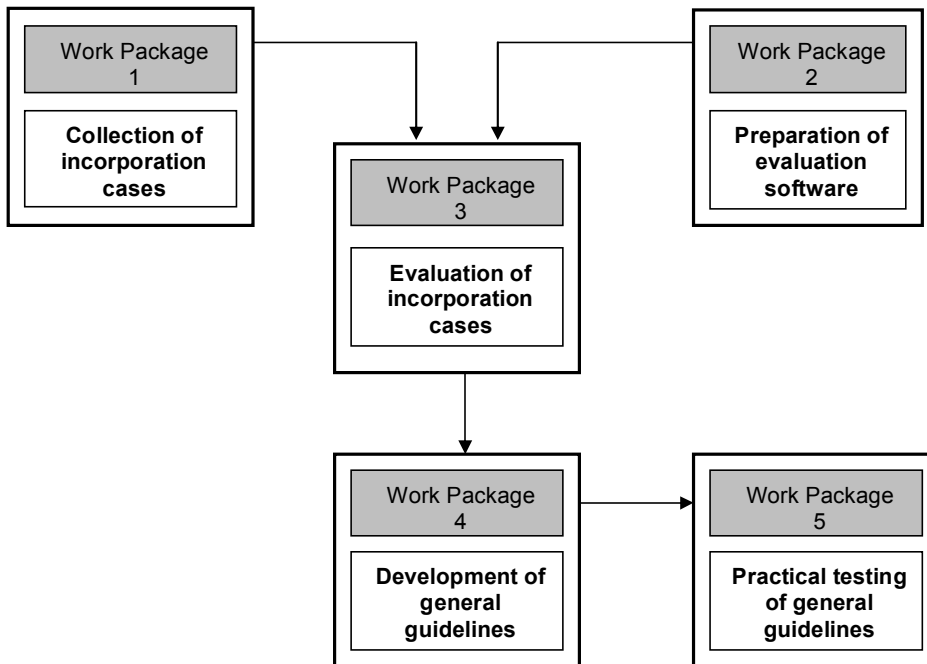


Figure 1. Structure of Work Packages

In Work Package 4, which is the core of the project (*Development of the general guidelines*), the partners derive a common strategy for the evaluation of monitoring data, draft the general guidelines and discuss it with internal dosimetry experts. The discussion will improve the common strategy and permit finalisation of the draft of the general guidelines.

In Work Package 5 (*Practical testing of general guidelines*) the validity of the draft guidelines will be tested by means of a dose assessment intercomparison exercise open to participants from all over the world (4<sup>th</sup> European Intercomparison Exercise on Internal Dose Assessment). The intercomparison will be prepared, all the participants will receive the guidelines and will be invited to use them during the assessment of incorporation cases.. The organisation of a Workshop (open to all the intercomparison participants), for discussing the results and finalising the report of the intercomparison is also scheduled. The last step of WP5 is the publication of the final version of the general guidelines and their submission to national and international bodies for approval.

## **State of the project**

The programme is running roughly according to the originally planned time schedule. At this time (August 2003) the following achievements can be reported.

Within WP1 two databases have been set up. The first database is a so-called Bibliographic Database, which collects information present in the open literature or in other reports dealing with internal contamination cases. All the participants in the project were involved in obtaining data from these and other sources of information. The structure of the database (BibDb) permits the user to view the database, search it and input new data. At this time more than 500 references have been collected. The references were distributed among the participants for reviewing and commenting on the papers from the point of view of their suitability for internal dose assessment (well documented cases). The selected case descriptions constituted the basis of another database called Internal Contamination Database (IntContDb). Besides the use of BibDb for the purpose of the IDEAS project, it also provides a useful tool for the scientific community interested in internal dosimetry for studying internal contamination cases. The BibDb has been put in a restricted web page presently available to the IDEAS partners only, but it will be made accessible for everybody in the near future.

The second database is the above mentioned Internal Contamination Database. This database has been set up to collate, in a given format, the descriptions of the selected well documented cases (contamination scenarios and follow up measurements). This means that the structure of the IntContDb permits the collection of all the information needed for internal dose assessment i.e. the description of the working area and characteristics of the work, date and modalities of the initiating event, actions taken, physical and chemical characteristics of the contaminant, etc.. For each contamination case, the participating partners entered the available information and monitoring data into a structured spreadsheet file for transfer into the database. Currently this database contains more than 200 cases. At present the IntContDb is accessible to the IDEAS partners only, but later will also be available on the web to others.

The IMIE (Individual Monitoring of the Internal Exposure) computer code was chosen for evaluation of the selected reference case studies. IMIE was developed for the purposes of retrospective dosimetry. It gives to the dosimetrist a very good tool for the analysis and interpretation of multiple bioassay measurements. IMIE helps the assessor to make estimations about a history of intakes and corresponding doses on the basis of individual monitoring data. In particular it permits the user to review and compare the possible variants of exposure condition combinations and to select the degree of automation from fully automated to completely manual regimes. Within WP2 the IMIE code has been improved and fitted to the special requirements of the IDEAS project. For instance during the course of WP2 a new optimisation algorithm of numerical deconvolution of monitoring data has been developed and a new probabilistic algorithm based on statistical methods has also been introduced. The final aim of WP2 has thus been achieved, namely to provide the participants a useful and flexible tool for the dose evaluation process of WP3.

After constructing the databases in WP1 and the upgraded IMIE code in WP2, the evaluation and analysis of the selected well documented cases was carried out in accordance with the scheduled work program of WP3. For this purpose 67 cases covering different circumstances and 16 radionuclides were selected from the IntContDb and distributed among the partners for detailed evaluation. The selected cases were evaluated using the IMIE and IMBA-Expert codes using different assumptions and making relevant comments. The best estimate for the calculated intake and committed effective dose were given in each case, together with notes on important issues related to the guidelines. The results were presented as MS Word documents and in



condensed version in Excel files in a given format and were collected in the evaluation database (EvalDb) established for this purpose. Up to now 75 independent evaluations on 42 cases have been collected in the database. The EvalDb provides possibilities, among others, to view the results of evaluations, to search within the database according to different aspects, to compare different evaluations on the same case and has links to the IntContDb. The results of the case evaluations have been discussed by the partners from many aspects (data fitting, uncertainties of data, intake patterns and model parameters, preferred monitoring types, handling of monitored data, etc.) and lessons drawn from the point of view of the guidelines to be prepared.

The tasks planned for WP4 have also been started by the preparation of the first outline of the IDEAS Guidelines Document that will harmonise with the corresponding ICRP document under preparation [3]. The draft of the IDEAS Guidelines is expected to be prepared by the end of 2003 and distributed for comments by means of a so-called Virtual Workshop to as many experts involved in internal dosimetry as possible all over the world. The Guidelines and the databases established in the IDEAS project will be put in the web page of the IDEAS project accessible to everybody, the address of which is for the time being

<http://hikwww2.fzk.de/hs/strahlenschutz/IDEAS/default.htm>

Comments will be invited and also posted on the Virtual Workshop web page up to the middle of March 2004. The comments will then be discussed by the consortium and a revised draft of the Guidelines prepared considering the outcome of the Virtual Workshop.

As scheduled in WP5 the validation of the Guidelines will be done through an intercomparison exercise, planned to be completed by the end of 2004, and followed by another Workshop to draw the final conclusions on the use of the draft Guidelines and to enable preparation of its final version.

### **Co-operations**

During the work of the IDEAS project close co-operation has been established with other EU projects (especially OMINEX [4]) and ICRP programs (Working Party on Bioassay Interpretation [3], Task Group on Internal Dosimetry, INDOS).

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# **Release Criteria and Management of Very Low Contaminated Scrap and Soil**

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## **Abstract**

Until 1996 no clearance levels were established in Slovak Republic; zoning principle was only used for release of potentially clean materials.

New supporting derived limits were issued in 2001 in accordance with IAEA /NEA guidance. Nuclides are grouped to 5 classes regarding their radiotoxicity.

European Commission recommendations were issued in 1998 for recycling of metals from the dismantling of nuclear installations. All relevant Slovak clearance levels are lower than those, recommended for EU.

## **Introduction**

There are currently more than a hundred nuclear reactors operating in the EU and around 40, many of which are research reactors, which have been shut down and are being decommissioned. During next decade significant amount of next NPPs will be shut down for decommissioning. This represents a large potential of material under regulatory control. A considerable fraction of this material, of which metals are the economically most valuable, is not or is only very slightly radioactive. Recycling or reusing this material would avoid unjustified allocation of resources to the disposal of low activity waste and save valuable natural resources.

The experience in Germany<sup>1</sup> shows, that total mass of radioactive components in the controlled area of large power reactors together with the buildings is 160 thousands tons (one unit PWR) or 230 thousands tons (one unit BWR). About 99% of radioactivity is fixed on 1% of mass (reactor pressure vessel and biological shield). More than 50 % of total mass is possible to release (free release) after cleaning and/ or measurement, next more than 30% is suitable for restricted recycling . Only less than 15% (mostly 5-10%) has to be declared and managed as radioactive waste.

The amount of cleared steel scrap in EU<sup>2</sup> in 2000-2015 is supposed to be in average 10 thousands tons per year, during next decade will reach value 30-40 thousands tons per year for massive dismantling of NPPs reaching the end of their designed life.

Slovenské elektrárne Inc. operates six units of WWER 440 type - 4 units in Jaslovské Bohunice and 2 units in Mochovce. They represents 32% of total Slovak installed capacity, nuclear share of electricity generated is yearly more than 50 %. First two units in Jaslovské Bohunice (NPP V1) passed principal reconstruction and safety improvements in 1998-2001 and are planned to be shut-down for decommissioning at 2006 and 2008 respectively. There is supposed, 63 thousands tons of metal scrap will be released after cleaning/ re-melting as well as 370 thousands tons of concrete suitable for recycling<sup>3</sup>.

Pilot nuclear power plant A1 was a HWGCR with channel type reactor KS 150 (refuelling during operation) and power output of 143 MW<sub>e</sub>. This NPP was operated from 1972 and was shutdown in 1977 after integrity accident of the primary coolant system. Significant damage and corrosion of the fuel cladding during the operational accident and

during spent fuel storage at NPP A1 caused consequently to existence of abnormal radioactive waste (containing alpha nuclides) and to the contamination of NPP parts. Extensive decommissioning works are carried out since 1995. There is supposed, 10 thousands tons of metal scrap if re-melting facility is constructed and used <sup>4</sup> and 180 thousands tons of concrete will be released during decommissioning.

#### Source of metal scrap cleared now and in the close future in Slovakia.

During refuelling on February 22, 1977, the non-conform fuel assembly (internal cross section was partly blocked by silica gel) was charged into the reactor core of NPP A1. Reduced coolant flow stream caused local overheating of fuel and consequent damage of technological channel and heavy water tank tube. As a result the loss of barriers integrity between fuel, carbon dioxide coolant and heavy water moderator took place. An extensive corrosion damage of fuel cladding by heavy water saturated with carbon dioxide occurred. The integrity of steam generator tubes was also influenced. The leakage of moderator and coolant mixture contaminated by fission products through corroded steam generator tubes resulted in the low contamination of the secondary circuit (turbine system). This material now represents the main volume prepared for decontamination and clearance.

Another source of very low-level radioactive metallic material was the reconstruction of NPP V1. The total amount 1500 m<sup>3</sup> of waste was produced, what represented more then 600 t of scrap and cables.

#### **Slovak legal basis for clearance and remediation**

Until 1996 no clearance levels were established in Slovak Republic; zoning principle was only used for release of potentially clean materials. Total radioactive contamination limits were 0.37 Bq/cm<sup>2</sup> beta, gamma and 0.037 Bq/cm<sup>2</sup> alpha without rules how to average values. These limits were established mainly for the purpose to remove non-contaminated equipment or other tools and items from control area.

The basic clearance principle issued in 1996 <sup>5</sup> is based on limit of effective dose for individual (an average member of the critical group) 10 micro Sv/y as well as collective effective dose 1 manSv/y from clearance acts. However, no material was cleared using this approach i. e. after the assessment provided for individual material and exempt act.

Application of the basic principle for metal scrap was not permitted, only supporting derived limits issued also in 1996 <sup>5</sup> for surface/specific activity could be used. They were as follows: 0,3 Bq/cm<sup>2</sup> beta, gamma, 0.03 Bq/cm<sup>2</sup> alpha and total specific activity 0.1 Bq/g. The clearance levels for conditional release (remelting of metal scrap with other non contaminated metal scrap) were also codified in 1996 <sup>5</sup>. They were based on fact of metal activity reduction (the radioactive nuclides are mostly remaining in slag) and dilution of remaining activity by melting with non- contaminated metal scrap, and allowed for remelting and consequent reuse the metallic scrap with the specific activity 10 x higher then required for direct release. The total activity of materials allowed for remelting from one site was less then 1 GBq/y. Since public was not prepared to accept such principle and facility for remelting metal scrap was not found, the above described "dilution approach" was later abandoned.

In 2001 <sup>6</sup> a new set of rules were issued for clearance and remediation in Slovakia. The basic clearance principle (based on limit of effective dose for individual 10 micro Sv/y and collective effective dose 1manSv/y from each exempt/ clearance act) became the general principle i. e. the alternative for release of such materials where the derived limits were also issued. It enables to release the material with contamination higher then derived limits if the way resulting to dose lower then dose limit is used.

New supporting Slovak derived limits were issued<sup>6</sup> in 2001 in accordance with IAEA guidance<sup>7,8</sup> on exemption/clearance principles based on individual nuclides activities (see Table I).

Table I  
Clearance levels for all types of cleared material<sup>6,7</sup>

Type of radioactive contamination	Radio-toxicity class				
	1	2	3	4	5
Materials, solid substances and items contaminated/activated through the whole volume	Clearance levels for specific activity [Bq.g <sup>-1</sup> ]				
	0.3	3	30	300	3 000
Materials and items contaminated on the surface	Clearance levels for surface activity [Bq.cm <sup>-2</sup> ]				
	0.3	3	30	3 00	3 000

Table II  
Radio-toxicity classes<sup>6,7</sup>

Class	Radio nuclides
1	Na-22, Na-24, Mn-54, Co-60, Zn-65, Nb-94, Ag-110m, Sb-124, Cs-134, Cs-137, Eu-152, Pb-210, Ra-226, Ra-228, Th-228, Th-230, Th-232, U-234, U-235, U-238, Np-237, Pu-239, Pu-240, Am-241, Cm-244
2	Co-58, Fe-59, Sr-90, Ru-106, In-111, I-131, Ir-192, Au-198, Po-210
3	Cr-51, Co-57, Tc-99m, I-123, I-125, I-129, Ce-144, Tl-201, Pu-241
4	C-14, P-32, Cl-36, Fe-55, Sr-89, Y-90, Tc-99, Cd-109
5	H-3, S-35, Ca-45, Ni-63, Pm-147

Nuclides are grouped to 5 classes regarding their radio-toxicity (Table II). A permission of Regulatory Authority under Ministry of Health (Radiation protection) for each type of release act is necessary (with exemption of individual small tools /items with surface less then 150 cm<sup>2</sup> and mass less then 10 kg). Measurement system including instrumentation and calculation of some nuclides activities based on nuclides vector must be justified and approved.

Activity is averaged over 1 t or 1m<sup>2</sup> for homogenous volume / surface activity and over 0.3 t or 0.1 m<sup>2</sup> for non-homogenous one. Non-homogenous activity can accede the levels given in table three times, general activity limit averaged over 1t or 1m<sup>2</sup> must be kept.

If material/items are contaminated in both volume and surface ways, the evidence that both limits are kept has to be done. If material/items are contaminated by more nuclides, sum of their aliquot portions cannot accede one.

In 2001, the legal basis for remediation of contaminated sites was established first time<sup>6</sup>. Any remediation activity needs a permission of the Regulatory Authority. The operator has to

provide the evidence, that the way of remediation is the optimal from the point of the view of radiation protection.

### Comparison of Slovak legal basis with European Commission recommendations

European Commission recommendations<sup>2</sup> were issued in 1998 for recycling of metals from the dismantling of nuclear installations. All relevant Slovak clearance levels are lower than those, recommended for EU even if the conservative approach for non-homogenous material and higher activity averaged through smaller surface is applied. Generally, the rules for averaging of surface contamination and volume activity in Slovak Republic are similar to those recommended for EU. No recommendation is given for EU concerning landfill release (concrete, soil).

In Slovakia there is no uranium mining, fresh fuel production or spent fuel reprocessing, therefore the most typical nuclides for release act from nuclear industry will be Co-60 and Cs-137 from NPP operation. A comparison of Slovak clearance levels for these nuclides with IAEA guidelines<sup>7</sup> and German approach<sup>9</sup> for clearance ranges for material from landfill, incineration, recycling and reuse shown, that Slovak values are so conservative, that only one set of clearance levels can be recommended for all kinds of cleared material. There are no derived clearance levels for sites in Slovakia. More details are given in Table III.

Table III

Comparison of material clearance levels (levels for sites are excluded)

Clearance levels	Mass specific activity (Bq/g)		Surface specific activity (Bq/cm <sup>2</sup> )	
	Co-60	Cs-137	Co-60	Cs-137
Slovak Republic <sup>6</sup>	0.3	0.3	0.3	0.3
EC <sup>2</sup> –metal scrap recycling (steel unrounded)	1 (0.58)	1 (0.58)	10 (9.1)	100 (39)
EC <sup>2</sup> -direct reuse (unrounded)			1(1)	10(3.7)
IAEA <sup>7</sup> – ranges for reuse, landfill, incineration, recycling	0.06-90	0.2-90		

Table IV Surface and mass specific clearance levels<sup>2</sup> resulting from IDE of 10 $\mu$ Sv/y for metal scrap

Nuclide	Surface specific clearance levels (Bq/cm <sup>2</sup> )		Mass specific clearance levels (Bq/g)
	scrap processing	direct reuse	steel scrap
H 3	3.7E+4	2.5E+4	1.4E+3
C 14	1.2E+3	7.7E+2	7.6E+1
Na 22	1.0E+1	1.1E+0	1.5E-1
Ca 45	2.9E+2	1.2E+3	5.8E+2
Mn 54	2.6E+1	3.7E+0	1.6E+0
Co 58	2.2E+1	8.0E+0	1.4E+0
Co 60	9.1E+0	1.0E+0	5.8E-1
Ni 59	1.5E+4	7.1E+3	3.8E+5
Ni 63	6.4E+3	3.0E+3	3.0E+5
Zr 95	9.2E+0	3.6E+0	9.0E-1
Ag 110m	8.3E+0	1.3E+0	5.1E-1
I 129	1.3E+1	4.0E+0	4.0E-1
Cs 134	1.4E+1	1.6E+0	2.1E-1
Cs 137	3.9E+1	3.7E+0	5.8E-1
U 238	5.9E-1	1.5E+0	3.7E+0
Pu 239	1.1E-1	2.6E-1	2.5E-1
Am 241	1.2E-1	3.1E-1	3.1E-1

The approach using basic clearance principle based on IDE 10 micro Sv/y allows to release under specific conditions metal scrap with higher activities. Table IV gives some examples. This approach recommended by EU, was not yet used in Slovakia.

### Recent clearance and remediation activities in Slovak Republic

The decommissioning of NPP A1 as well as the reconstruction of NPP V1 have been recently the main sources of low contaminated scrap and concrete.

During NPP A1 operation and later on, as a result of significant contamination of NPP A1 systems, waste collection and storage tanks; the soil around concrete tanks and pipelines was contaminated. As a consequence of waste tanks overflow during flooding also soil in bottom and on the banks of liquid discharges and rainwater channel from NPP area was contaminated. Although the contamination of soils out of NPP site is very small and from the point of the view of radiation protection no significant measures are necessary, local authorities under the pressure of public required to remove some contaminated soils and add them to low contaminated soils and concrete stored at the NPP site.

### Inventory and release of scrap from decommissioning and refurbishment of NPPs

Since no requirements for measurement (with exemption of regular calibration) were established before 1996, any surface contamination measurement was accepted as the evidence that activity level was under limit. The only materials from non-active parts and systems of NPP A1 were released in 80s after 100% measurement of surface. Two pieces of

metallic supporting beams, each 7.5 t were reconstructed after blasting and reused for new spent fuel management equipment.

More than 1200 t of steel scrap resulting from dismantling of low contaminated auxiliary equipment and secondary circuit of NPP A1 (in 80s) were continuously sorted and prepared for decontamination. Total estimated activity is  $10 \text{ E}+11 \text{ Bq}$ , what represents average specific activity  $120 \text{ Bq/g}$  (mostly Cs -137). After commissioning of decontamination facility with sufficient capacity, 100% measurement using qualified instrumentation was required for both beta, gamma and alpha contamination and for surface and/or specific activity. Lack of equipment enabling efficient measurement of such a low especially specific activities caused, that no material after decontamination was released.

Another source of very low-level radioactive metallic material was the reconstruction of NPP V1. The total amount  $1500 \text{ m}^3$  of waste was produced, what represented more than 600 t of scrap and cables. After qualified measurement of 100% surface, 238 t of scrap and 112 t of cables were released. The rest will be processed at Treatment and Conditioning Centre. The remaining material will be released or conditioned and disposed after fragmentation, decontamination and measurement. Only a few non-contaminated concrete crushes were released from NPP V1 reconstruction of non-active systems after measurement giving evidence of no contamination.

After 2001, material release was interrupted; three new equipment and relevant procedures for both surface and mass specific activities measurement were implemented and are used starting 2002-3.

Materials from NPP V1 reconstruction, NPP A1 decommissioning and Interim Spent Fuel Storage reconstruction will be released after decontamination and residual activity control. During Interim Spent Fuel Storage reconstruction, more than 300 t of steel structures will be changed, decontaminated and measured for release till 2010.

### **Contaminated soil and concrete**

The total volume of  $12\,500 \text{ m}^3$  of contaminated soil was collected in the past during reconstruction and decommissioning works mostly on NPP A1. The soil was sorted according specific activity and is now stored at different storage places at the NPP A1 site.  $4000 \text{ m}^3$  of soil with specific activity  $0.5\text{-}3 \text{ Bq/g}$ , was mixed with soil with lower contamination (the total volume is  $9600 \text{ m}^3$  with average activity  $0.35 \text{ Bq/g}$ .) in concrete trenches (former cooling tower basement structure), covered by inactive soil and grassed-grown. This soil was after safety assessment released from regulatory control (left at the place with the possibility of institutional control).

Large amount of contaminated soils is foreseen to be collected during forthcoming decommissioning and remediation operations at the site:

- about  $4\,000 \text{ m}^3$  of contaminated soil which will be collected during 1<sup>st</sup> phase of NPP A1 decommissioning; total estimated Cs-137 activity is  $2 \cdot 10^{10} \text{ Bq}$

- about  $3\,000 \text{ m}^3$  of contaminated soil supposed to be collected during next phases of NPP A1 decommissioning

- $1\,000\text{-}2\,000 \text{ m}^3$  of river bottom sands and sediments

- cca  $3\,000 \text{ m}^3$  of contaminated soil which will be collected during decommissioning of NPP V1 and NPP V2

Landfill disposal at the site for total volume of 20 000 - 30 000 m<sup>3</sup> (approximately 100m x 60m) of very low contaminated soil is under discussion and design preparation. The main reasons to place the landfill at the NPP site are economical, public acceptance, well-known geological, hydro geological, seismic and meteorological situation and possibility of simple institutional control (common site for NPP A1, NPP V1, NPP V2 and Centralised waste processing facility).

## **Conclusions**

A great effort was paid during the last decade to develop a strategy, legal basis and corresponding rules with the aim to establish proper conditions for implementation of safe and sound clearance and release procedures.

Slovak legal basis and general approaches have been harmonised with European Commission and IAEA approach and recommendations; in some cases, the Slovak rules are more restrictive.

Some practical examples of implementation are given, as well as closest perspective activities in this subject are indicated. Most of them are directly related to decommissioning of NPPs A1 and V1 in Jaslovské Bohunice.

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# NUCLEAR ENERGY AND ENVIRONMENT IN CLOSE FUTURE

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Nuclear energy is evidently the best energy hope of mankind. It represents the way how to decrease the contamination of atmosphere and it solves partially the problem of global increase of temperature. Nuclear energy prevents of 1.8 billions tons of emissions. In Europe it represents 550 mil t of CO<sub>2</sub> emissions yearly. This is equivalent of 140 mil cars.

The nuclear industry in US produces about 20 % of electric energy and in 2001 it was the first time when the production of nuclear energy was cheaper than that from fossil fuel (coal). The power plant V1 in Jaslovske Bohunice is the second most cheapest source in our country, producing energy for half of the average expences of our all energy sources.

We can be sure that the nuclear energy will be the nondispensable part of energy sources also in future. Nuclear energy owns a vast innovation potential. Further research and development have to sensitively react on the needs of acceptability, compatibility and sustainability of nuclear energy. The safety of nuclear energy, radioactive waste deposition, and prevention of risk from nuclear materials misuse have to be very seriously abjudged and solved.

The main aims of nuclear energy are:

- to continue to keep an important share in energy production in long term perspective,
- to innovate and improve technology and safety of nuclear industry,
- permanently improve its economic, environmental and mental compatibility and acceptability.

WETO (World Energy Trading Organisations) tried to suggest the description of the future world energy system.<sup>4</sup> According to their conclusions, world energy demand is projected to increase at about 1.8%/year in the next three decades. While industrialised countries experience a slowdown in the growth of their energy demand 0.4 %/year in the EU, conversely, the energy demand of developing countries grows rapidly. The world energy system will continue to be dominated by fossil fuels almost 90% of total energy supply in 2030. Oil will remain the main source of energy (34%) followed by coal (28%). Natural gas will represent about 25 % of world energy supply. Coal demand could grow rapidly as it becomes more competitive than other fuels. The change in the fossil fuel shares impacts considerably on the carbon intensity of the world energy system and on the associated CO<sub>2</sub> emissions.

Due to the continued dominance of fossil fuels, world CO<sub>2</sub> emissions are expected to increase more rapidly than the energy consumption (2.1%/year on average), reaching in 2030 twice the level of 1990.

Nuclear energy increases slightly in absolute terms. During the 1990-2000 decade the growth of nuclear was 2.7%/year, but this rate weakens to 0.9%/year over the projection period. In 2030, nuclear represents 5% of the world GIC, compared to 7% in 2000.

Globally, energy from renewable sources is expected to cover 8% of world energy requirements in 2030 essentially due to the continuous decline of traditional biomass consumption in the Third World.

The development of nuclear power does not keep pace with total electricity production: nuclear world market share comes down to 10 % of total electricity production in 2030. World electricity production from renewables is expected to rise from 2 % in 2000 to 4 % in 2030, mainly because of a rapid increase in the electricity production from wind.

Mainly the two type of nuclear power plant can be considered for close future<sup>4</sup>:

- standard large Light Water Reactor (LWR)
- new evolutionary nuclear design.

Two kinds of blocs attract attention of nuclear community nowadays.

- It is **EPR**, the improved and verified reactor PWR with many original elements of inherent safety, developed in cooperation of France and Germany (Framatom, EdF-CNEN, Siemens KWU) and
- pressurized reactor cooled and moderated with light water advancing CIS row of blocs, **VVER-640/V407** with 1800 MWt and 640 MWe (contemporary 290 MWt).

The basic difference among these two groups of blocs is according to attitude to the largest reactor incident. While VVER640 concentrates to prevent by all means the melting of central reactor zone improving and suggesting new technologies of cooling, the EPR concept is based on the fact, that when some safety systems will fail and the central zone will be melted, the fused mixture will flow down to the protected position, where it will be partially under control.

In the frame of international research also some other projects will take place in the first half of this century:<sup>2</sup>

- light water cooled systems – innovation for deeply burnt fuel, energetic fission of actinides, concept of supercritical reactor, etc.,
- gas cooled systems – high-temperature thermal reactor, modular HTR, and two industrial prototypes of GT-MHR and PBMR prospective till 2010, fast neutron gas cooled reactor, etc.,
- special cooling media – concept with melted salts, liquid metals, especially Na, Pb and Pb-Bi-eutectics,
- cogeneration – combination electric production – desalted water,
- hydrogen production – hydrogen economics and hydrogen - petrochemical market.

Standard large Light Water Reactor (LWR) is supposed to exhibit capital costs slightly increasing over time due to increased investment in security measures. In the technology case, the investments as well as O&M costs are assumed to be about 35 % lower as compared to 2030.

New evolutionary nuclear design. This technology is assumed to be introduced gradually after 2010 and costs about 30 % less to construct than the LWR by to 2030 thanks primarily to its inherent safety characteristics. Probably it gain a substantial share of the total nuclear market (approx. 12 %). For the nuclear technology case this type of plant is assumed to be 35 % cheaper to construct and 35 % to operate.

The overall effect of nuclear technology case is a worldwide reduction of CO<sub>2</sub> emissions in 2030 of 2.8% (4.6 % in the OECD). At global scale there is expected considerable increase in nuclear electricity generation. Overall nuclear contribution can increase from 9 % to over 15.5 %

(from 16 % to 37 % in the OECD). Nuclear power can penetrate into the high to medium annual loads displacing coal and gas fired electricity production.

Nuclear power plants (NPP) are developing permanently. They represent the most satisfactory part of energy and electric sources. WANO<sup>3</sup> (World Association of Nuclear Operators) has pointed the progressive attributes improving their techno-economic characteristics:

Unit Capability Factor has raised during the last 10 y from 77 % to 83 % representing equivalent of new 45 reactor blocs given into operation

Collective Radiation Exposure felt down during the last ten years approximately to 1.1 ManSievert/bloc

Coefficient of Industrial Safety Rate has decreased during the last ten years to one half of previous value.

NPP grow older. About 25 % of reactor blocs are working for more than 25 y. The key moment for today's nuclear energy is to receive the permission to operate for further period. Only this can prevent 10 % decrease in nucleus expected for 2010 y in US. After deep complex control of blocs, Oldbury (Magnox) asked for 10 y prolongation of the permission to operation to 40 y, Calder Hall asked for 10 y to 50 y, Oconee, Hutch and others asked the same in US, Tokai (Kansai Electric) finally asked for 30 y of prolongation to 60 y of operation.

According to the last trends the attitude of broad European community to nuclear energy changes. The majority votes for atomic power plants (e.g. in US 60 %, in South Korea 92 %, France 67 %, Germany 81 % etc.) Convention from Kyoto on reduction of CO<sub>2</sub> emissions supports the case.

Today the content of CO<sub>2</sub> in atmosphere increases about 1 %/y. Since 1860 y the highest mean Earth temperatures are measured these years. And to fulfill obligations from Kyoto seems difficult. Japan, to reach the aim (-6 %) ought to construct 20 new 1300 MWn.blocs till 2010 y. In EU (-8 %) the rate of decrease is slower than expected. US (-7 %) refuse to reduce emissions according to Kyoto at all, pointing that it could threaten their further economic development. Not negligible is also the need of developing countries for energy. New reactor blocs construction is shifted to Asia, where the increase of share of nuclear sources will reach additional 8 % to total 26 %! South Korea plans next 10 blocs, North Korea 2 blocs, Taiwan 4 and Kina 6 new blocs.

Also new technologies help to support nucleus. New generation of reactor blocs is projected. In US the National Environmental and Engineering Laboratory (INEEL), together with the Argonne National Laboratory create the Top Center of R&D in nuclear energy mainly in the field of advanced nuclear technologies, GENERATION IV and technology of advanced nuclear fuel cycle. Around 2030 they could start new era of advanced nuclear energy blocs with higher safety, reduced wastes and better economy. In US the Final Design Approval for advanced project AP 600 got Westinghouse and for boiling reactor ABWR it received GE (General Electric). In France, European Pressurized Reactor is preferred. Its economic advantage can be seen at contracted construction of at least 6 – 7 blocs. One billion FF has been already invested into R&D. Two pressurized reactors are already working at Penly and Seine-Maritime, Fr.

More close communication of operators with inhabitants is accented to dissolve their negative expectations. The possibilities of effective bloc shut down and its final liquidation, as well as the routes for high radioactive waste processing are emerged. It can be demonstrated by projects of liquidation of NPP Tokai-1 (250 mil. USD, 15 y), NPP Zion-1, NPP Vandellós-1 (300 mil. USD, 30 y), etc.

Nuclear Power activities can enhance also liberalisation of Energy Market. In US full liberalisation, in EU at least 25 % according to low. Problems can be expected in case of reconstruction works in progress.

Important effects on nucleus may impose energy producers:<sup>3</sup>

- a) To increase load factor by:
  - the shortage of reactor general repairs
  - the decrease of nonplanned intervals
- b) To increase the nominal output of reactor bloc
  - Loviisa (Fin) – already the increase about 100 MW (11 %)
  - Olkiluoto (Fin) – the increase 125 MW (pile 710 MW/bloc), 2 blocs
  - NPPs in Spain – 500 MW increase together in next ten years
- c) Exclusion of nucleus from electricity corporations and unite them into great nuclear conglomerates with more united operation and better support.
- d) Internalisation of nuclear energy production of electric energy :
  - Comp. Amergen goes to buy NPS TMI + NPS Oyster Creek
  - Comp. British Energy goes to invest into Ontario Hydro
  - Comp. EDF invests to Hungary, Austria, Swiss nuclear energy marketsTake place also the broad fusion process: Siemens AG + BNFL (British Nuclear Fuel PLC), Magnox + BNFL, IVO + NESTE, Framatome + Cogema, Tractatel + Iberdrola, etc.

Renewables. The main technologies affected by R&D probably improved in their techno-economic characteristics during the next decades could be:

- Biomass gasification for electricity production in small scale (less than 25 MW) combined cycle plants
- Photovoltaics
- Molten Salt Tower Solar plant with storage. This technology will be realised in decades after 2030 in developing countries where most of the physical potential exists.
- Small hydro, assumed to be a mature technology registering insignificant gains over the projection period.
- On-shore wind turbines (over 500 kW capacity), highly competitive, in spite of its intermittent character, with massive development worldwide but without significant impact to electric market.

The Renewable results in a 3 % reduction in worldwide CO<sub>2</sub> emissions.

The technology cases defined here do not offer definitive solutions for the global CO<sub>2</sub> emission problem. This is largely because the power generation sector represents only a part of the energy market. The extension of these energy technology cases to other important CO<sub>2</sub> emitting sectors (road transport, the residential and tertiary sector) should be a priority in the future.

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**IMPLEMENTATION OF NEW POLICY AND PRINCIPLES OF  
HARMONISATION OF NUCLEAR EMERGENCY  
PREPAREDNESS IN CONDITIONS OF EMERGENCY  
RESPONSE CENTRE OF THE NUCLEAR REGULATORY  
AUTHORITY OF THE SLOVAK REPUBLIC**

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**1. Introduction**

With respect to Chernobyl accident the changes in understanding of nuclear emergency preparedness have initiated a developing process resulting in an effective enhancement of conditions ensuring adequate response to nuclear or radiological accidents of emergency situations in many countries. The Slovak Nuclear Regulatory Authority (UJD) in frame of co-operations with IAEA, EC, OECD/NEA and other international organisations has actively participated in this challenging work targeting implementation of international experience and best practices in the country. The new international policy (principles declared e.g. in ‘Memorandum of Understanding’, IAEA, Vienna, 1997) based on experiences propagating importance of regional co-operation, harmonised approach and clear strategy for protective measures implementation in case of a nuclear or radiological accident has influenced the development also in Slovakia. The implementation process in the country was supported by changes in legal conditions regulating peaceful use of nuclear energy [1,2] including basic rules for emergency preparedness published in the second half of 1990 years.

**2. Role of the Emergency Response Centre of the Nuclear Regulatory Authority of the Slovak Republic in nuclear emergency response preparedness**

With respect to legal conditions [1] the Slovak Nuclear Regulatory Authority has established its own Emergency Response Centre (ERC) to have sufficient tools and means for an independent assessment of emergency situations and to be able to provide professional

recommendations to the authorities at national level. Legal, administrative and technical support enabling work of ERC have been implemented in compliance with principles of international approaches and at the same time taking into account national conditions for emergency management. Several projects of technical co-operation (IAEA, EC, bilateral with UK, US, Japan etc.) significantly accelerated this development and in the second half of 1990 the ERC was presented as a capable system supporting the nuclear emergency response in the country. The ERC has been integrated into the National Emergency Plan as a technical support centre for National Emergency Commission for Radiation Accidents (NECRA) and in direct and close co-operation with the Technical Advisory Committee of NECRA it is understood as a primary source of technical information for decision-making process during an emergency situation at national level. This position of the ERC enables to provide advices to the plant region on the development of the accident and on the countermeasures protecting the public through the NECRA. In case of serious severe accident situations ERC could provide an independent assessment also directly to co-ordinating local governments.

From technical and administrative point ERC also ensure the operation of international contact point (CP) with respect to conventions [3,4] signed by member states of IAEA and it is planned to provide a platform for integration with the notification system ECURIE of the European Union. With respect to EC project RODOS the ERC integrating the National Centre of RODOS system in the country should be part of this European system enabling prompt exchange of the data and prognosis results over the Europe region.

### **3. Administrative, technical and personal conditions of ERC**

ERC uses modern technologies to keep the communication uninterrupted – keeping a fax and phone contact point 24 hours a day and 7 days in a week. The CP (also IAEA official CP) enables both domestic (including nuclear facilities) and international exchange of information. The on-line data transmission systems in real-time mode ensure effective acquisition of data necessary for ERC teams expert work. Transmitted data have been selected with respect to requirements of ERC emergency procedures and computer-based support systems. The volume of transmitted data is about one hundred technological data per unit (minutes sampling time), a few ten of data characterising radiation situation and local meteorological situation at the site (5 minutes sampling time) and broader meteorological (regional and Europe) data. The existing operational data acquisition system collecting technological and radiation monitoring data extended by local meteorological information, data from large area monitoring systems and meteorological networks increases the reliability

of results of computer based support systems in ERC. Legal, administrative and technical support enabling operation of ERC have been implemented in compliance with typical requirements implemented in several developed countries (e.g. France, Finland etc.). Basic conditions for a response in case of nuclear or radiological accidents are described in set of procedures for ERC. These procedures cover all functional responsibilities of the ERC and have been developed to support all activities of ERC staff and ERC operation. Besides of generic and organisational part, professional parts deal with all specific areas necessary to give comprehensive results on accident development. These documents include sufficient conservativeness and take into account availability of limited information (technological, radiological and in case of off-site impact also meteorological data) in case of an accident. The most important professional procedures are included in part of (III) Reactor Safety, (IV) Source Term, (V) Exposure and public radiation protection, (VI) Monitoring parts. The procedures have been developed on base of calculations (sequences) simulating severe accidents (and related events) using validated codes for VVER units (MARCH, MAAP4/VVER, MELCOR, RELAP etc.) and were mostly done by VUJE company.

The existing emergency plan, valid emergency procedures based on pre-calculated accident sequences and installed dynamic code packages as SESAME/VVER, ADAM, RTARC and the European RODOS system guarantee the credibility of processed information and reliability of results prepared for decision making process. With respect to operational modes of the ERC the activities are divided in three basic levels:

1. Monitoring level
2. Emergency level
3. Post-emergency level.

For all modes the ERC permanent staff has to fulfil various tasks necessary for verification of the correct functionality of the ERC equipment and procedures and at the same time has to maintain the staff capability to perform activities needed at any time. The basic structure of the ERC team is divided with respect to functional requirements into the following groups:

1. The Reactor safety group (RSG) is responsible for monitoring the status of the reactor plant and assessing for the possible accident development
2. The Radiation protection group (RPG) is assessing current radiological situation including the impacts of the releases of radioactivity on workers and members of the public

3. The Information group (IG) is coordinating the technical briefing material prepared by the RSG and Radiation protection group. It is also responsible for international information exchange
4. The Logistic group keep running the ERC and provides necessary support to other groups.

#### **4. Regional and international co-operation and harmonization in the field of emergency preparedness**

One of the most important features of modern trends of emergency preparedness is the regional harmonization. Particularly in case of small countries, as it is typical in Central Europe and in Europe region, regional co-operation should be of a high importance. Effective communication and information exchange has been recognised as a key element of emergency plans. With respect to international conventions [3,4] and good relationships with neighbouring countries the Slovak Republic initiated and signed bilateral agreements with many countries to support pillars of a common co-operation in case of nuclear or radiological emergencies and strengthening co-operation in the field of preparedness. With respect to IAEA RER/9/050 and RER/9/064 common national trainers courses on IAEA methodology for emergency preparedness [5,6] and [7] were organised enabling sharing of experience between Slovak and Czech experts. Based on experience in the area several workshops were organised to present results and to share experience with countries from Eastern and Central European region (Armenia, Lithuania etc.). The Slovak Nuclear Regulatory Authority has been also active in hosting fellowship and scientific visitors from countries, where arrangements for emergency preparedness are under development. With respect to good co-operation with neighbouring countries a few proposals ('Trilateral exercise' – Slovakia, Hungary, Austria, Poland; 'Vysehrad proposal' - Slovakia, Hungary, Czech Republic, Poland) were prepared and agreed to improve the level of co-operation. These projects had to enable validation and comparison of national arrangements, including national plans for nuclear or radiological accidents, emergency procedures and support systems formally integrated in national arrangements, as well as testing of effectiveness of bilateral agreements between countries. These projects also had to contribute to the enhancement of public opinion and had to demonstrate progressive governmental policy and interest to increase the nuclear safety in the region.





## 6. Conclusions

The principles of emergency preparedness in Slovakia fully support regional harmonisation and co-operation. Effective implementation of international practice and sharing of experience substantially contributed to the level of emergency response in the country and to the harmonisation of emergency response preparedness creating also conditions for an efficient regional integration.

[1] Act on peaceful use of nuclear energy 130/1998 Collection of laws

[2] Decree on emergency preparedness 245/1999 Collection of laws

[3] Convention on Early Notification of a Nuclear Accident, IAEA, Vienna 1987

[4] Convention on Assistance in the Case of a Nuclear Accident or Radiological Emergency, IAEA, Vienna, 1987

[5] Method for the development of emergency response preparedness for nuclear or radiological accidents, IAEA-TECDOC-953, 1997

[6] Generic Assessment procedures for determining protective actions during a reactor accident, IAEA-TECDOC-955, 1997

[7] Generic procedures for monitoring in a nuclear or radiological emergency IAEA-TECDOC-1092, 2000

# **RADON PROGRAMME IN THE CZECH REPUBLIC**

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## **Introduction**

The framework of the Radon programme in the Czech republic includes both precautionary measures and interventions. The programme informally started in early eighties has been now incorporated in national legislation (Atomic Act, Radiation Protection Decree, etc.). Aim of precautionary measures is to avert construction of building above natural radiation guidance levels (200 Bq/m<sup>3</sup> for indoor radon concentration and 0,5 µSv/h for gamma dose rate) by protection of new buildings against soil radon ingress, by regulation of natural radioactivity in building materials and supplied water. Aim of interventions is to identify buildings affected by enhanced natural radioactivity and help owners to put into effect reasonable remedial measures. Two sets of intervention levels for indoor natural exposure were established: guidance intervention levels 400 Bq/m<sup>3</sup> (indoor radon), 1,0 µSv/h (indoor gamma dose rate) and limit values 4000 Bq/m<sup>3</sup> and 10 µSv/h.

The radon programme is based both on governmental and private activities. The governmental activities include representative and targeted indoor radon survey, subsidy for radon mitigation, mitigation test measurements and public information on radon issue. The private activities include radon measurement (radon index of building site, indoor measurements, radon diagnosis) and remedial measures. More than 100 commercial companies were authorised by Radiation Protection Authority (SUJB) to provide these measurements.

## **The system of precautionary measures**

Precautionary measures cover monitoring and control of all potential radon sources: soil gas, building material and supplied water. The procedure is as follows: Estimation of “radon index” of building site is obligatory during siting of a new building. Radon index category - low, medium or high - is determined by set of radon in soil gas measurements (15 probes in depth 0,8m) and permeability measurements in the building site. It is not accepted to use radon geological prognosis maps of the territory, the maps are generally not detailed enough for such purposes, even in scale 1:50 000. It is obligatory to protect buildings against soil

radon in compliance with technical code taking into account radon index category of building site. Essential characteristics of different types of building materials - e.g. radon diffusion coefficient of radon proof membranes - are tested by standard procedure. It is evident however, that further research of “leakage area”, “joints” and sealing materials ought to be carried out yet.

Monitoring and regulation of natural radioactivity of building materials are the second important precautionary measure. All producers of building materials are obliged to monitor content of  $K^{40}$ ,  $Ra^{226}$  and  $Th^{232}$ , evaluate I („mass activity index“) <sup>(1)</sup>

$$I = \frac{C_{Ra}}{300 \text{ Bq} \cdot \text{kg}^{-1}} + \frac{C_{Th}}{200 \text{ Bq} \cdot \text{kg}^{-1}} + \frac{C_K}{3000 \text{ Bq} \cdot \text{kg}^{-1}}$$

and submit results of measurements to SUJB. Exemption level  $I = 0.5$  was accepted for building material used in bulk amount. Producer must carry out optimisation of radiological protection and cost benefit analysis, if exemption level is exceeded. Limit level 150 Bq/kg  $Ra^{226}$  for building material used in bulk amount was set to limit radon exhalation.

Monitoring and regulation of natural radioactivity in supplied water are the last precautionary measure. Suppliers of water into public water distribution networks are required to ensure systematic measurement and evaluation of the content of natural radionuclides ( $Rn^{222}$ , gross alpha, gross beta). If exemption levels (50 Bq/l -  $Rn^{222}$ , 0.2 Bq/l -gross alpha, 0.5 Bq/l -gross beta) are exceeded then content of natural radionuclides must be analysed and optimisation/cost benefit analysis of possible remedial measures must be carried out. Limit values set for individual natural radionuclides cannot be exceeded in any case. There is no obligatory limit for individual water sources, values mentioned above are used only as guidance levels.

The survey of effectiveness of precautionary system was carried out in the last year. It was found out surprisingly, that indoor radon level 200 Bq/m<sup>3</sup> was exceeded somewhere in about 20 % of new houses. One of the reasons seems to be unexpectedly low air exchange rate in modern energy-saving houses.



on new insulation, sub soil ventilation, computer controlled pressure air-exchange rate etc. Tests of mitigation efficiency are carried out in two steps: 1) after mitigation short-term measurements (at least one week) by commercial companies, 2) long term measurement (1 year) by track detectors provided by SURO. Malfunctions of remedial measures were investigated by SURO expert group. Unfortunately, the research of effectiveness had shown that nearly 25 % remedial measures realised within past ten years had failed from long term point of view. The new governmental radon mitigation subsidy is more restrictive now. It should be provided subsequently and only if mitigation was really successful.

### **Information and public awareness**

The radon bulletin and special leaflets are periodically prepared to improve public awareness of radon issue. They are distributed to all Construction offices and to mayors of municipalities in radon prone areas. Special leaflets are prepared on preventive measures and remedial measures. The quantitative survey on radon awareness among Czech republic residents carried out 2 years ago showed relatively good knowledge on this topic. The question “Have you ever heard of naturally occurring gas named radon?” was answered: “Yes, I know it very well” (16 % of respondents), “I have heard something” (59%), “No” (25%).

### **Conclusions**

There are some issues important from standpoint of practice and interesting from scientific point of view those could be topic for further research:

the right indoor radon measurements and estimation, investigation of variability of short/long term measurements, influence of confounding factors (meteorology etc), development of new radon diagnostic methods to identify and quantify radon sources and entry rates, investigation radon permeation through the building construction, investigation of validity of radon mapping process in details, relation indoor radon vs. soil gas radon, influence of geology parameters and building technology, investigation of new mitigation methods, long term effectiveness, the role of energy saving buildings.

### **References**

1. Radiological protection principles concerning the natural radioactivity of building materials, EU Commission: Radiation Protection 112, 1999
2. Barnet I. Mikšová J., Procházka J. Radon database and radon risk map 1:500 000 of the Czech republic. The four international workshop on the Geological Aspects of Radon Risk mapping. Prague 1998

# DEVELOPMENT OF RADIATION PROTECTION IN SLOVAKIA

*Petrášová Mária*

*Slovak Nuclear Society - WIN Slovakia*

Radiation protection is a discipline of science, which includes many areas of knowledge such as medicine, engineering, technology, law and others. Therefore it was difficult for me to decide about the content of my presentation.

First of all I would like to emphasize that the history of radiation protection in our country is as old as the discovery of X ray radiation and radioactivity. To be aware of radiation hazard, the State Radiological Institute in the first Czechoslovak Republic was established in 1919 in Prague. The activities dealt with in the institute were mainly the measurements of natural radioactivity and implementation of appropriate standards for the quantities and units of ionizing sources. The founder of the institute was the world famous scientist Professor Frantisek Behounek whose professional knowledge and dynamic approach to the issues of radiation risk established a basis for the radiation protection as a scientific discipline.

I will not speak about the history of radiation protection before the Second World War. It would be interesting, but because of the lack of information, I would only like to remind the pioneering work of Prof. Behounek. I have to say that the development and history of the Slovak and Czech radiation protection have gone in parallel. Since 1918 till 1992, we had lived in a common state – the Czechoslovak Republic - where the single radiation protection policy had been applied.

After the Second World War – due to the discovery of nuclear fission and later to nuclear weapon tests - increasing attention was paid to radiation protection.

In 1952 on the basis of the Act No. 4/52 on Health protection, the Institutes of Occupational Medicine and the Institutes of Industrial Hygiene and Occupational Diseases were established in Prague and Bratislava. Simultaneously regional and district hygienic stations were created to deal with urgent practical routine tasks, first of all in the area of occupational hygiene. After the government decision on the construction of nuclear power plants in Slovakia in 1955, departments of radiation protection were built in the institutes.

The main tasks of the research institutes were focused on experimental studies related to radiotoxicological problems of the most important radionuclides. The laboratory of Health Physics in the Bratislava-based institute began to investigate the radiation burden to bodies of patients and personnel during special radiodiagnostic procedures. Later on the research was focused on qualitative and quantitative analyses of alpha, beta, gamma and neutron radiation, on investigation of the content of natural and man-made radionuclides in environmental samples. The institutes have also ensured methodical leadership of radiation protection departments in hygienic stations, which were successively established at regional levels.

In fifties, many X ray diagnostic units for medical applications were in operation. Most of them were in bad technical conditions. Therefore the main tasks of the radiation protection departments in the regional hygienic stations were to the removal of existing technical defects of the units and to harmonize the activities of these units with the then legislation. Besides these activities, local radiation protection problems had to be addressed. These problems included the radiation protection of miners and environmental monitoring. The content of natural and man-made radioactivity in drinking water, in lakes and rivers was investigated and the monitoring of man-made radioactivity in food, in forage and soil was carried out. Long-term studies of radioactivity fallout have been carried out since 1959. An interesting study on the content of  $^{90}\text{Sr}$  in teeth and bones of children and adults in Slovakia was developed in sixties after intensive nuclear weapon tests in the atmosphere.

An expert group was brought together in the radiation protection department in the regional station of hygiene for Western Slovakia (Bratislava-based) for addressing problems of radiation protection during the design, construction and licensing of the first NPP A1 in Jaslovské Bohunice. At that time it was a very complex subject, which needed to develop national radiation safety standards. An advisory committee managing the standard publishing and the decision-making process for nuclear industry was thus established. The main objectives of the committee actions were to ensure the radiation protection of workers and the general public living near nuclear power plants.

In spite of the fact, that the new ICRP recommendations had already been published (ICRP 26 /1977 ICRP 60 1991) they were not included in our legislation. It should be stressed that in cooperation with the IAEA and Russian specialists, as well as with the help of hardly available scientific reviews, we were able to apply the conclusions of the new ICRP recommendations for the regulatory control in radiation practice. At that time Chief Hygienist instructions were put in force. They were related to the regulations of radiation protection



supervision, population dose and radioactive waste management. The instructions assisted in and made easier our decision-making.

Several important changes in the Slovak radiation protection legislation were made after 1989. The general Health Protection Act No. 272/1994 was published and amended by the Acts No. 290/1996 and 471/2000. The Act introduces the basic subjects of radiation protection, dose limits and main responsibilities, obligations and duties to the utilization of ionizing radiation. These principles and limits are compatible with international standards. To implement these acts, the Regulation of the Ministry of Health No.12/2001 on the requirements for providing radiation protection was put in force recently.

The extensive activities in radiation protection were done during and after the Chernobyl accident. The accident brought a wide spectrum of radionuclides that were deposited also on our territory. The evaluation of radiological situation required an intensive monitoring. Immediately after the accident, Czechoslovak Radiation Monitoring Network for monitoring and decision-making was created in Slovakia and Bohemia, too. We surveyed the sufficient amount of experimental data to assess the potential risk to the public. The results of the investigation created the basis for decisions about measures how to mitigate accident consequences. I should underline that only the participation and cooperation of many scientific and university institutions together with the radiation protection departments on plant sites and with the radiation protection departments in the regional hygienic stations made it possible to assess the radiation hazard for the population.

In addition to the Chernobyl accident we had to face other radiation events or emergency situations. I would like to mention some of them.

The first radiation event that we had to address was an extensive contamination with radioactive luminescent paints. The paints contained radionuclides  $^{90}\text{Sr}$  and  $^{226}\text{Ra}$ . Contaminations of working places, workers households and internal contaminations of personnel were found. The personnel were not informed about the radiation risk of their activities. The working places and households had to be decontaminated. The decontamination was controlled and supervised by the regulatory body for radiation protection. The decontamination lasted more than 4 weeks.

The next accident we were engaged in was an aircraft crash with all 82 passengers killed, which occurred near Bratislava in 1966. A consignment containing  $^{131}\text{I}$  with the total activity of 83.25 GBq was on its board. The aircraft was destroyed by the collision and fire and the containers were broken contaminating the environment. Our measurements demonstrated high local surface soil contamination with the materials transported. The

decontamination had to be done until the surface contamination decreased less than 37 kBq on the effective area of 150 cm<sup>2</sup>. Permission of the regulatory body for radiation protection had to be given for the post mortem examination and for the transport of victims.

It would be difficult not to mention the radiation accidents at the Bohunice A1 plant. The first event occurred in January 1996 when the cooling medium CO<sub>2</sub> contaminated with fission and corrosion products leaked out into work places and then in plant surroundings. In this event 2 persons were killed as a result of suffocation by CO<sub>2</sub>. The radiological consequences for the environment and the general public were evaluated as negligible.

The second event in February 1967, which initiated due to overheating and damage of fuel rods, brought the Bohunice A1 plant into decommissioning state. A significant leakage of radioactive materials in the hydrosphere was recorded.

Abnormal rainfall on Bohunice A1 plant site and insufficient measures against flooding led to a flooding of rooms in the plant controlled area in June 1978. A huge amount of contaminated water was produced. The contaminated water subsequently had to be released into the recipient of Dudváh river and then to Váh river. Notwithstanding the increased radioactivity of the effluents, no immediately countermeasures for the mitigation of consequences had to be done. Water in these rivers is used for irrigation of fields. And at the time of those events in January 1997 and in June 1978, no irrigation was needed.

At the end of my lecture I would like to note that the health effects of radiation are relatively well understood and can be effectively minimized through careful safety measures and practices. I believe that the development of radiation protection in cooperation with the ICRP, IRPA, IAEA and other international organizations will successfully continue in Slovakia and that we have to achieve one of most important aims - to ensure that radiation sources and radioactive materials are managed for both maximum safety and human benefits and that they assist in achieving a sustainable life for future generations.

# **SOURCES OF IONIZING RADIATION, RADIOACTIVE OR NUCLEAR MATERIALS OUT OF CONTROL. NATIONAL SYSTEM OF RESPONSE IN SLOVAKIA**

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## **Introduction**

During last 20 years many incidents have been reported world wide about accidentally melted radioactive materials or radioactive materials out of control. Regarding the risk associated with possible accident and the wide range of consequences, the responsible authorities have taken measures to avoid the health risks due to possible external irradiation or internal contamination, both for the members of the public and the possibly involved workers. The total number of sources respectively workplaces, mainly in the field of industry was decreasing slowly during last ten years. The decommissioning became very expensive and so where it was possible, the technology equipped with radiation sources was upgraded by measuring technology based on other principles (e.g. ultrasound).

In many cases the radioactive materials - wastes result from old installations or practices which lost the authorization or have never been authorized. Such radiation spent sources have been found in areas of hospitals (radium needles), process industries (old gauges) and research institutes where these spent radiation sources had been used in the past. In most cases the present owner is going bankrupt and has not money to finance the returning of the spent source to the supplier (if exist/is known) or the authorized decommissioning. These sources present a potential risk because such facilities are often closed and are not a subject to regular inspections.

In the case that the radioactive material/source is of unknown origin, in accordance with §17r clause 10 of the Slovak Act No. 272/1994 in the wording of amendments “dealing with institutional radioactive waste of unknown origin is allowed only to legal person authorized by the Ministry of Health. The Ministry will determine the legal person possessing such authorization to deal with radioactive waste of unknown origin in the form of written decision.” The costs in that case are paid by the state. At present the authorized legal person to deal with the radioactive materials of unknown origin is the company HUMA LAB APEKO Ltd.

### **Customs inspection - present situation**

The activities of the Slovak Customs Service in the field of non-proliferation nuclear and radioactive materials are as follows:

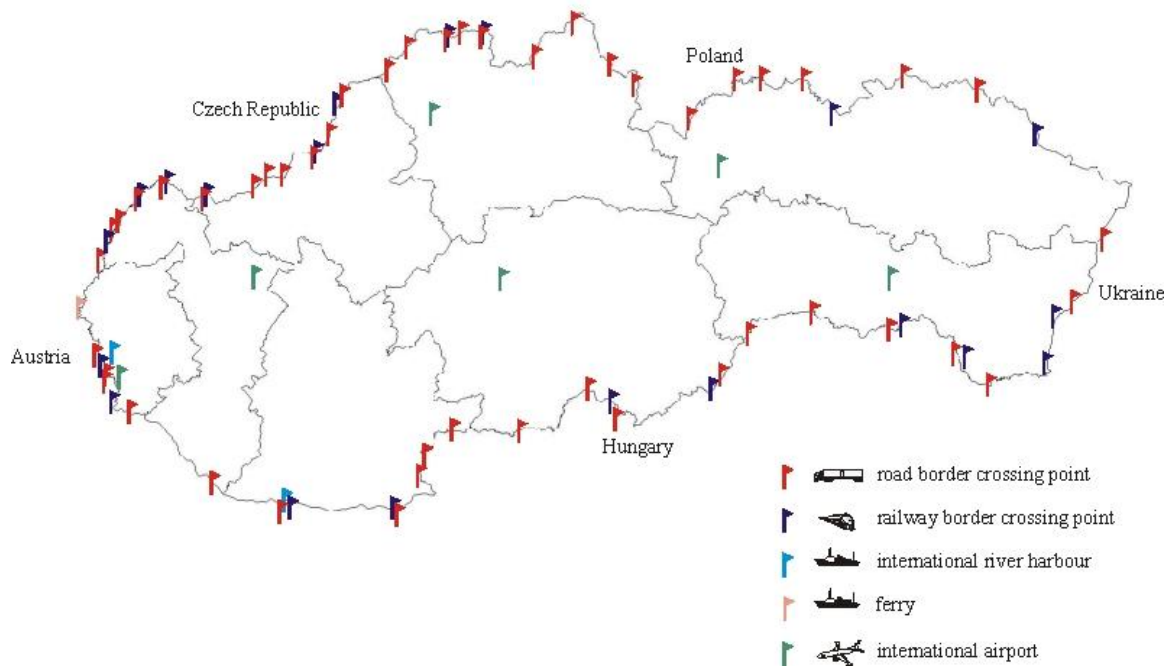
- Measurements at the border crossing points (see. Fig. 1)
- Selection of suitable detection equipments and a development of internal regulations
- Delivery of the training for customs officers
- Cooperation with other governmental and non-governmental bodies
- Enforcement activities and intelligence

Stationary detection systems are used for the permanent monitoring of vehicles on the entry side of the border crossing point to the territory of Slovakia - 4 systems at the State border between Slovakia and Ukraine and 2 systems at the State border between Slovakia and Poland. Additional equipment consists of handy detection units (10 pieces of Eberline FH 40 G, 40 pieces of TSA PRM 470 A, handy detection units, 3 pieces of Isotope Identifiers GR - 135 MiniSPEC) and PAGERS which are used by the Special Forces of the Slovak Customs during special investigation operations.

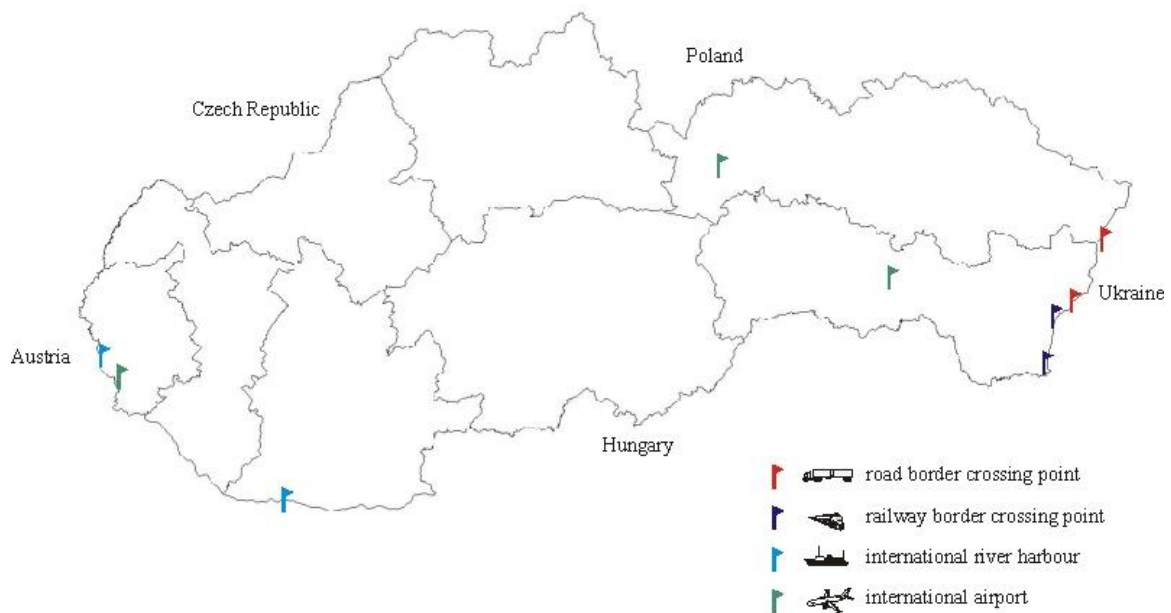
### **Customs inspection - situation after accession process to European Union**

- Measurements of the goods will be logged off by Slovak Customs at the border crossing points with Czech Republic, Poland, Hungary and Austria;
- Customs inspection will be carried out just on:
  - 3 international airports (Bratislava, Poprad and Košice)
  - 2 road border crossing points (Vyšné Nemecké and Ubl'a)
  - 2 railway border crossing points (Čierna nad Tisou and Maťovské Vojkovce) at the state border between Slovakia and Ukraine – the new external EU border (see Fig. 2)
  - 2 international river harbours (Bratislava, Komárno)

**Fig.1 Customs inspected border crossing points  
Present situation**

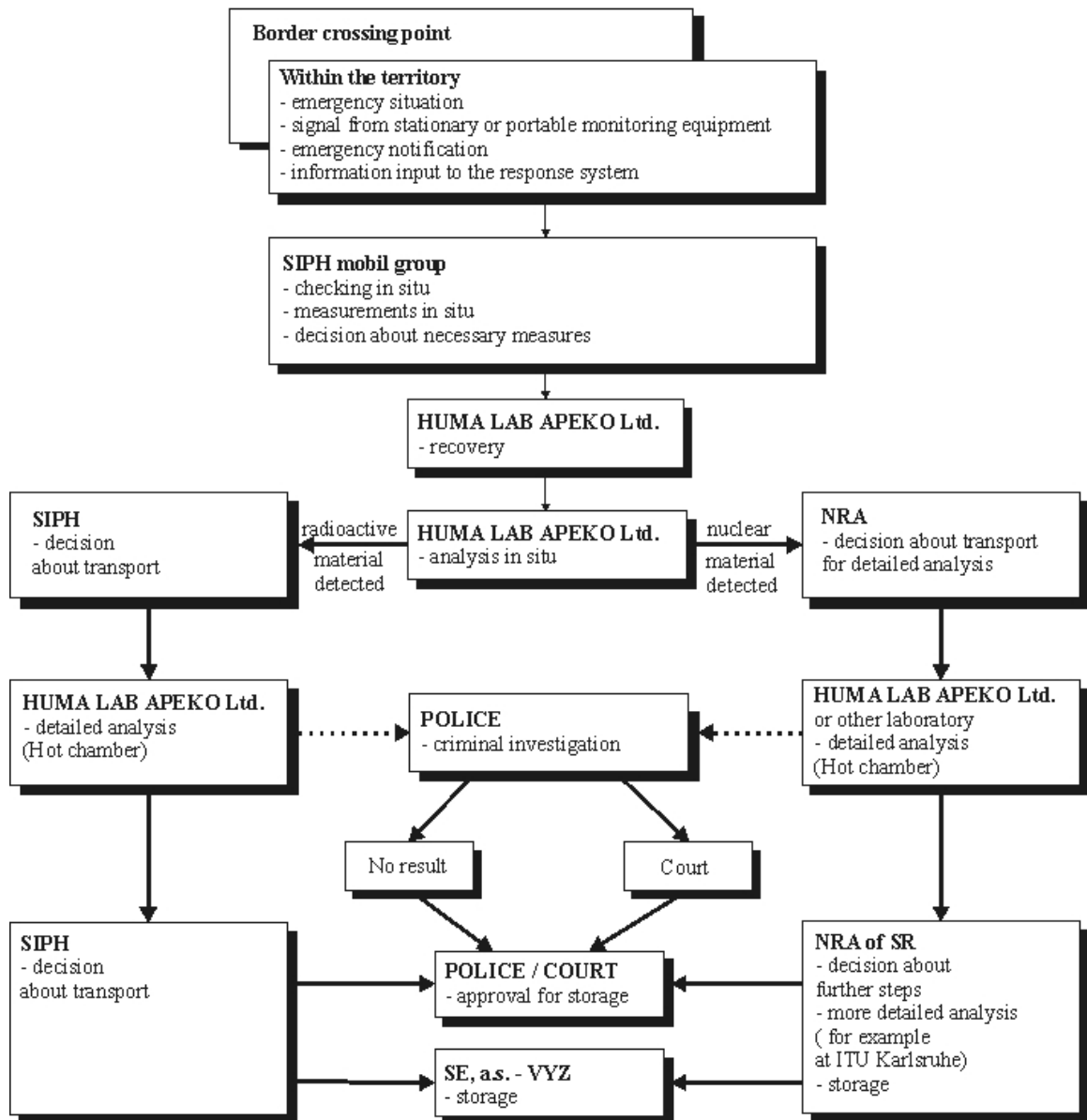


**Fig.2 Customs inspected border crossing points  
after accession process to EU**



In the field of combating the smuggling of nuclear and radioactive materials the Slovak Customs Administration has focused its activity to strengthen the new external EU border (to force training of the staff, to deliver the latest equipment, to improve the response plan etc.)

## Response scheme



The actual response system to incidents with orphan sources or radioactive material occurring in metal scrap, illicit trafficking and disused sources out of control is laid down on the following scheme. The national strategy is aimed to establish a more effective responding system preventing further illegal trafficking with regard to the acceding process which will require for new member states joining EU proper arrangements in improving the safety of radiation sources over the life-cycle to ensure the effective functioning in the conditions of the Slovak Republic's membership in the European Union.

# DOSIMETRY SERVICE IN SLOVAK REPUBLIC

*Zuzana Koreňová*  
*Slovak Legal Metrology, Bratislava, Slovak Republic*

## **History:**

1995 – Dosimetry service was established by authority Institute for Standardisation, Metrology and Testing of Slovak Republic  
1996 – The Laboratory of Personal Dosimetry at Slovak Metrology Institute began its activity  
1998 – transformation to the Department of Personal Dosimetry which became a part of Slovak Legal Metrology, metrological workplace Bratislava

## **Monitoring of individual doses**

Department of Personal Dosimetry provides personal dosimetry service of external exposure for all occupational workers except them of two Nuclear Power Plants in Jaslovské Bohunice and Mochovce.

Number of regularly monitored persons: about 7000 from medical, industrial, research and other areas for about 400 organisations.

Number of evaluated doseimeters: about 45 000 per year (80% from medical field).

Dosimetry service maintains also dose histories of about 15 000 persons.

The data will be retained until age 75, not less than 30 years after termination of work with sources of ionising radiation.

The dose reports are sending monthly to the organisations.

In the case of exceeding the limit is always phoned to the organisation and written a message to the Regional State Health Institute.

## **Dosimetry System:**

TLDs

Equipment: two TLD Readers HARSHAW 6600

Radiation Evaluation and Management System (REMS)

Personal doseimeters: whole body  
extremities (finger, wrist)

TL chip: LiF:Mg,Ti (TL100, TL600, TL700)

Holders: type 8814, 8805, ring, bracelet

Measured quantities: personal dose equivalents  $H_p(10)$ ,  $H_p(0,07)$ ,  $H_p(3)$

Dose range: 0,1 mSv ÷ 1,0 Sv

Frequency of evaluation: 1 month, 3 month (quarterly)

## **Regulations:**

Act No. 272 / 1994 on the Protection of Human Health amended by the act No. 290 / 1996 and No. 470 / 2000

Health Protection against Radiation Decree No. 12 / 2001.

## **Exposure:**

Exposure limits:

workers: effective dose 50 mSv per year

effective dose 100 mSv per 5 consecutive years

equivalent dose 150 mSv lens of eye

equivalent dose 500 mSv skin, extremities

students (16 – 18 years) – participating in professional training programmes:

effective dose 6 mSv per year

equivalent dose 50 mSv lens of eye

equivalent dose 150 mSv skin, extremities

## **System of quality:**

Slovak Legal Metrology ( the part of which is Department of Personal Dosimetry ) was:

accredited according to STN EN ISO / IEC 17 025

certificated according to EN ISO 9001: 2000.

The Department of Personal Dosimetry is licensed by the Ministry of Health for monitoring of Personal Doses – External Exposure.

## **QC and QA:**

Before each measurements:

check of environmental conditions (temperature, humidity, gas flow)

check of operational parameters

check of the reader correction factor (RCF)

Once per month:

preventive maintenance of readers

Once per half-year:

check of the calibration of the system

Once per year:

calibration of the system

verification of the system and evaluation of dose quantities

(The Slovak Institute of Metrology)

Internal audit (STN EN ISO / IEC 17 025, EN ISO 9001:2000)

Other:

interlaboratory comparisons:

Intercomparison of Radiation Dosimeters for Individual Monitoring (1996 – (1998)

(IAEA Vienna)

Intercomparison for Individual Monitoring of Radiological Measurements for Monitoring Purposes, Personal Dose Equivalent  $H_p(10)$  (1999 – 2000)



(IAEA Vienna)

External audits in accreditation (The Slovak National Accreditation Service)  
in certification (CE Qualite Slovakia accredited certification body for certification of  
management systems Nová Dubnica)

### **Attachments**

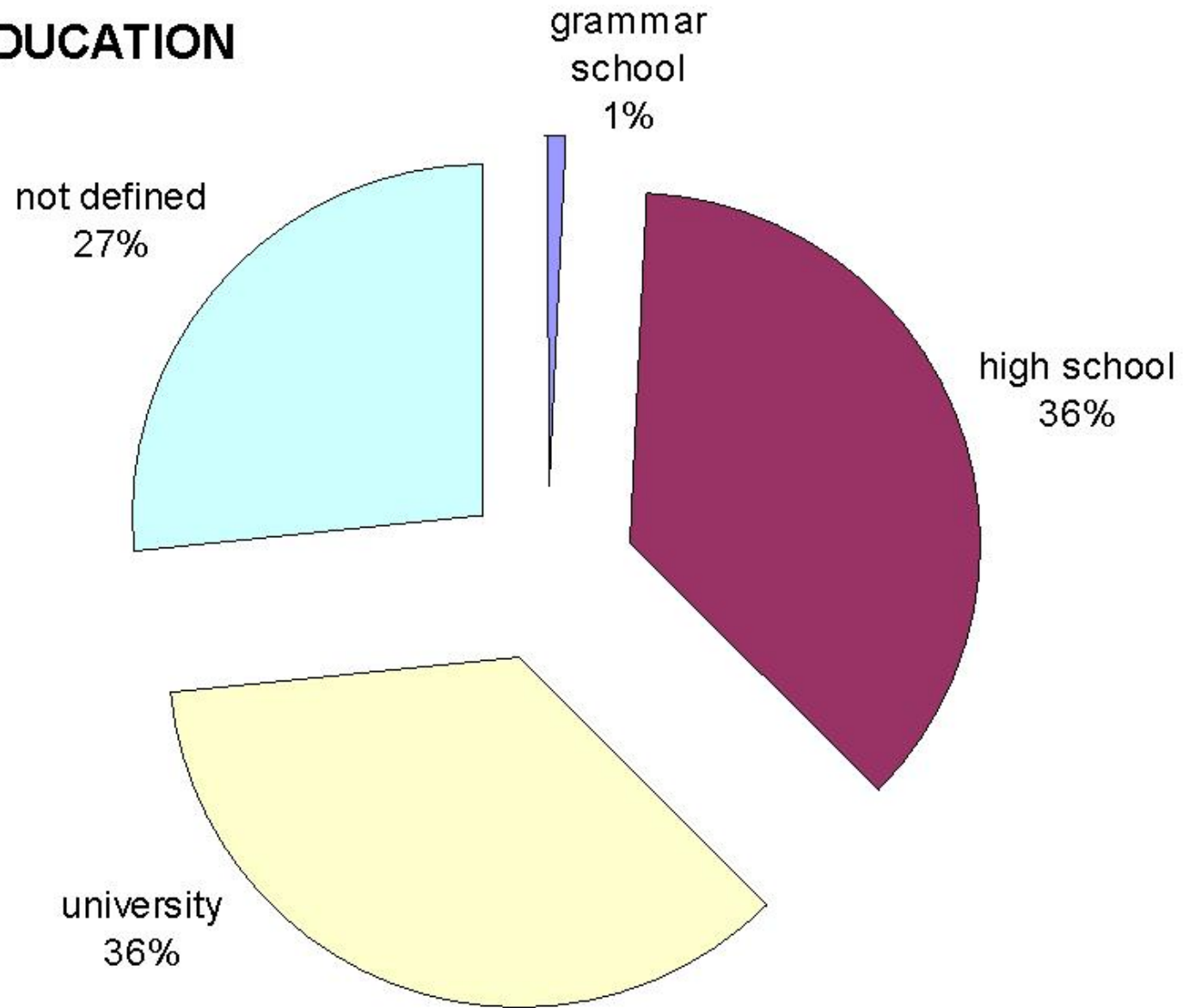
Graphs survey of education of monitored workers  
in detail high schools and universities  
frequency of monitoring  
survey of work places (occupation)  
in detail health service (work sector medicine)  
and specialised work places

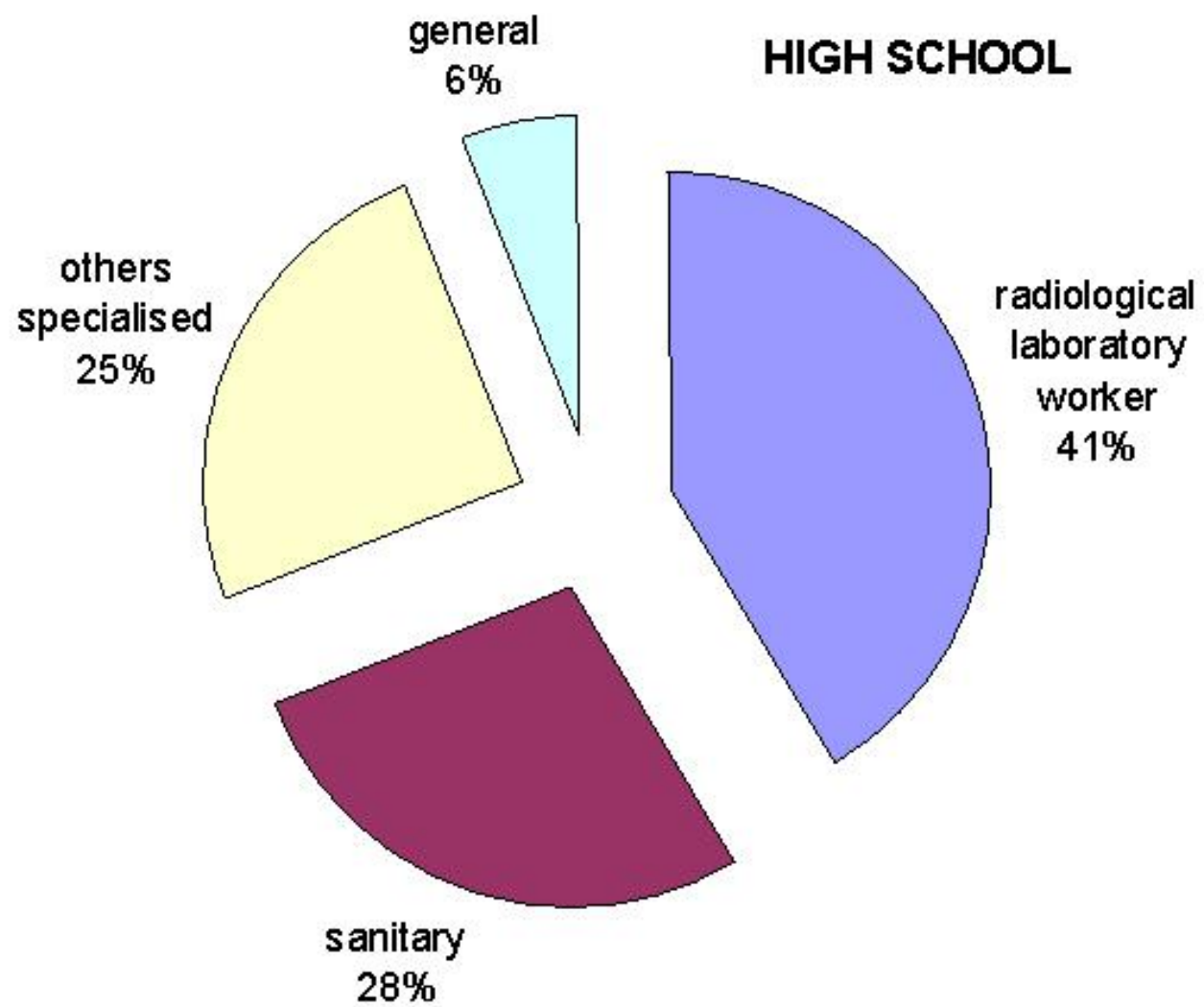
Table Data survey on the numbers and doses of occupationally exposed persons  
for the year 2002

Acknowledgment to:

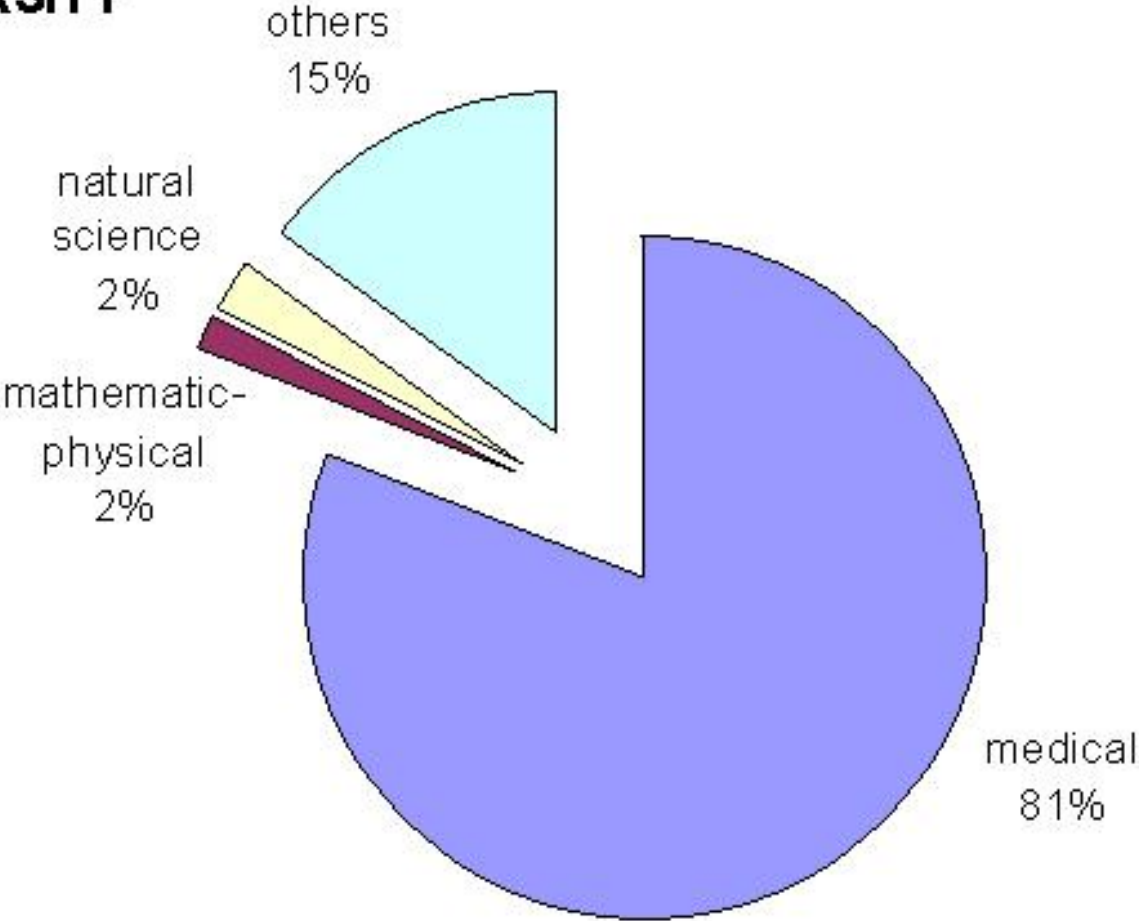
Monika Koreňová and Martin Koreň for their help with preparing and data processing  
To my husband for his patience

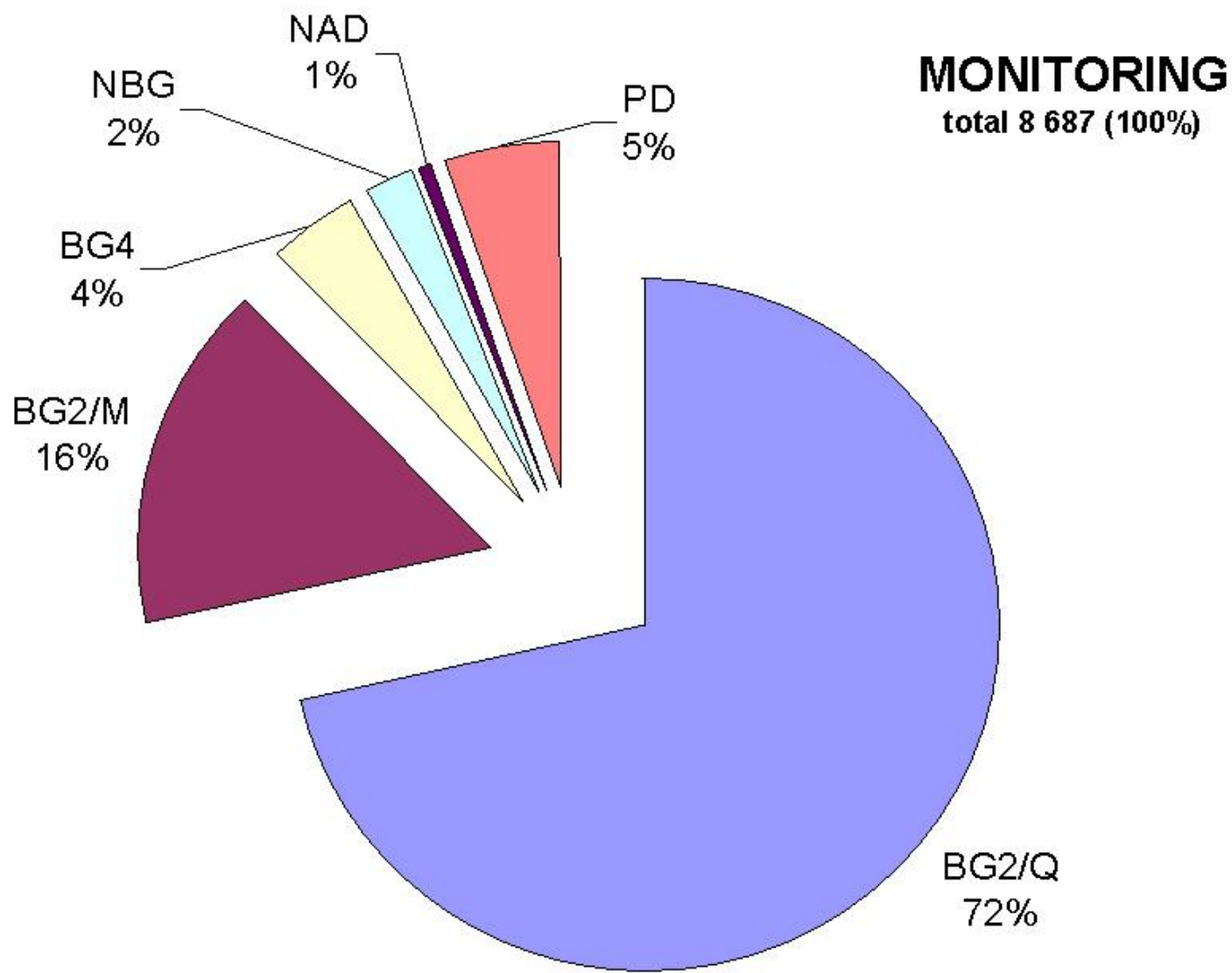
# EDUCATION



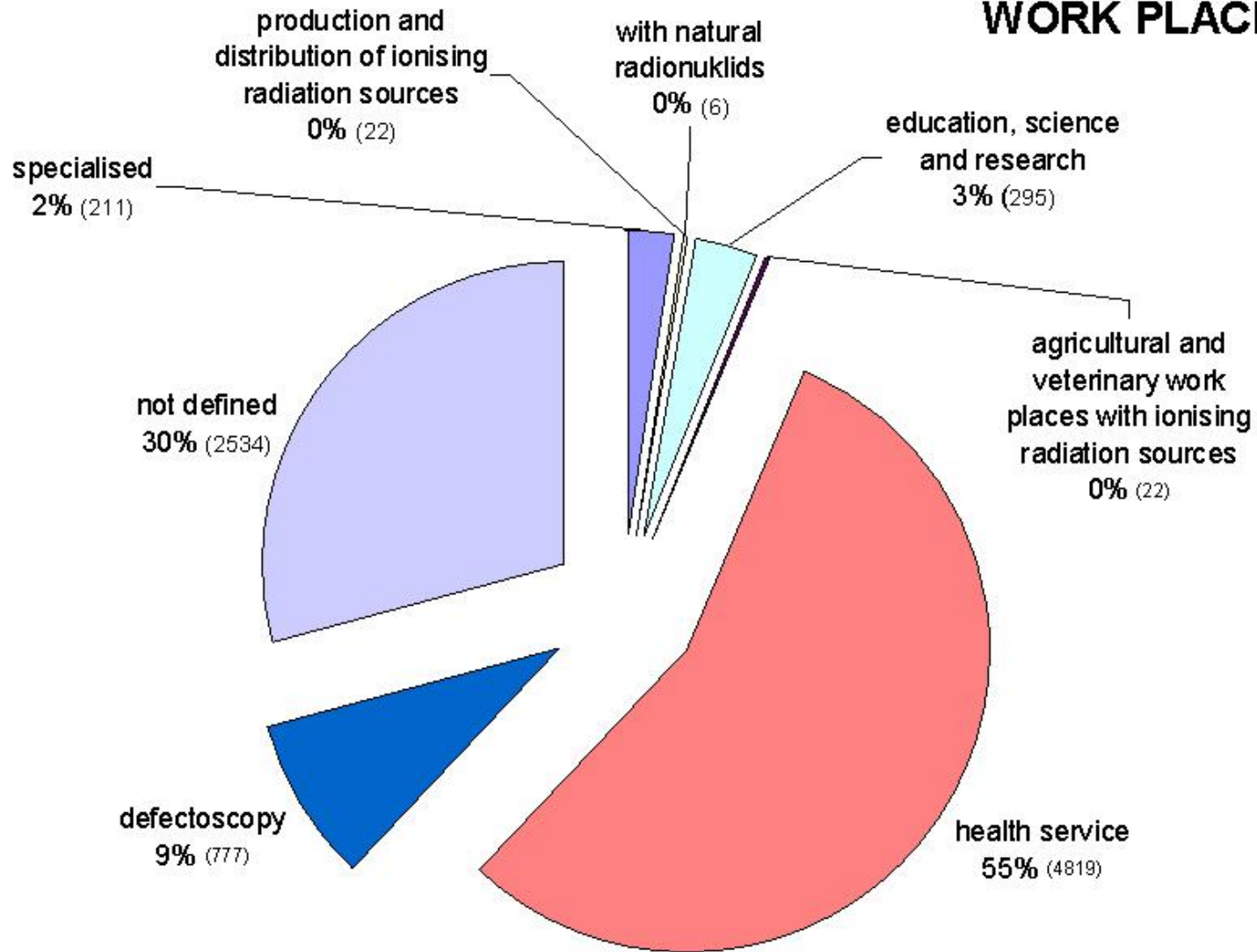


**UNIVERSITY**

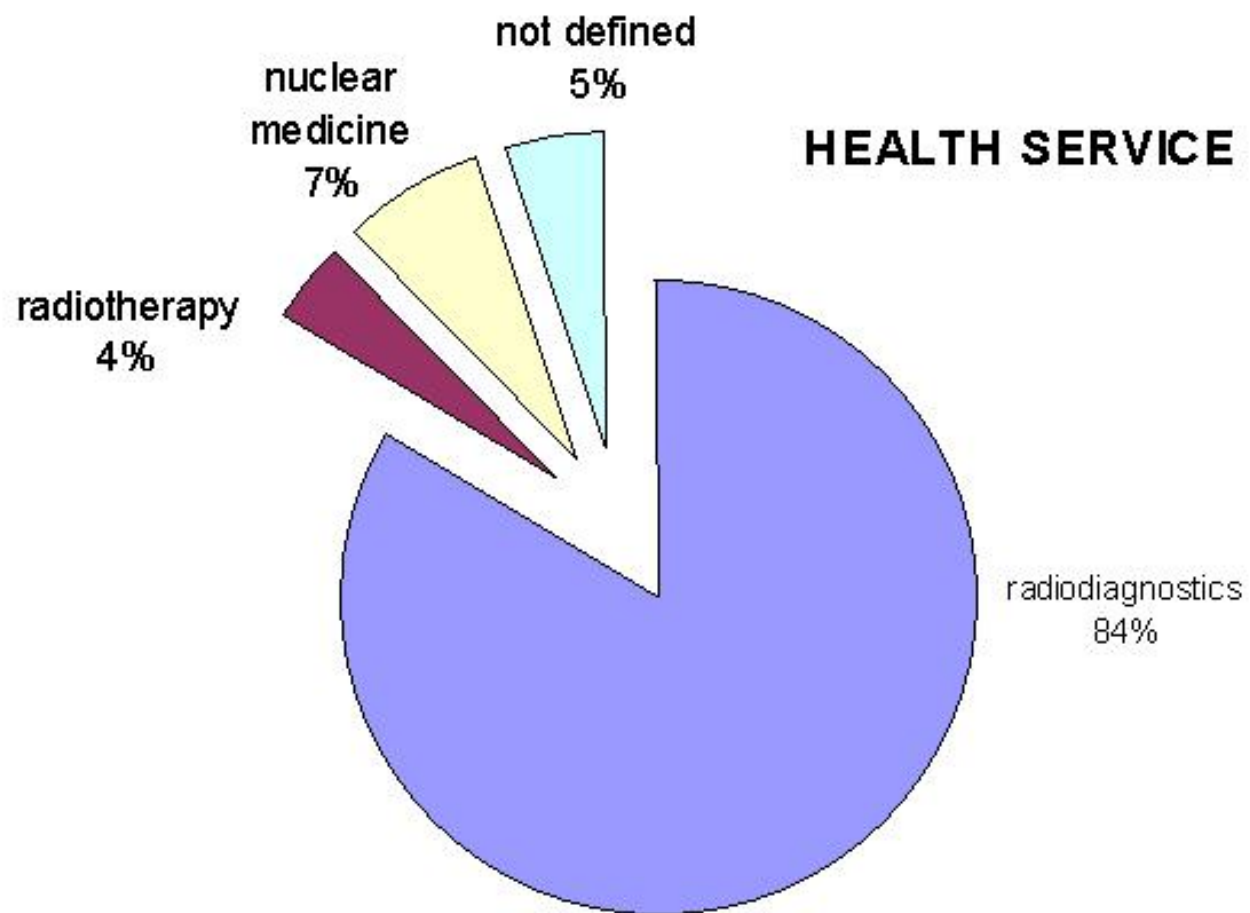




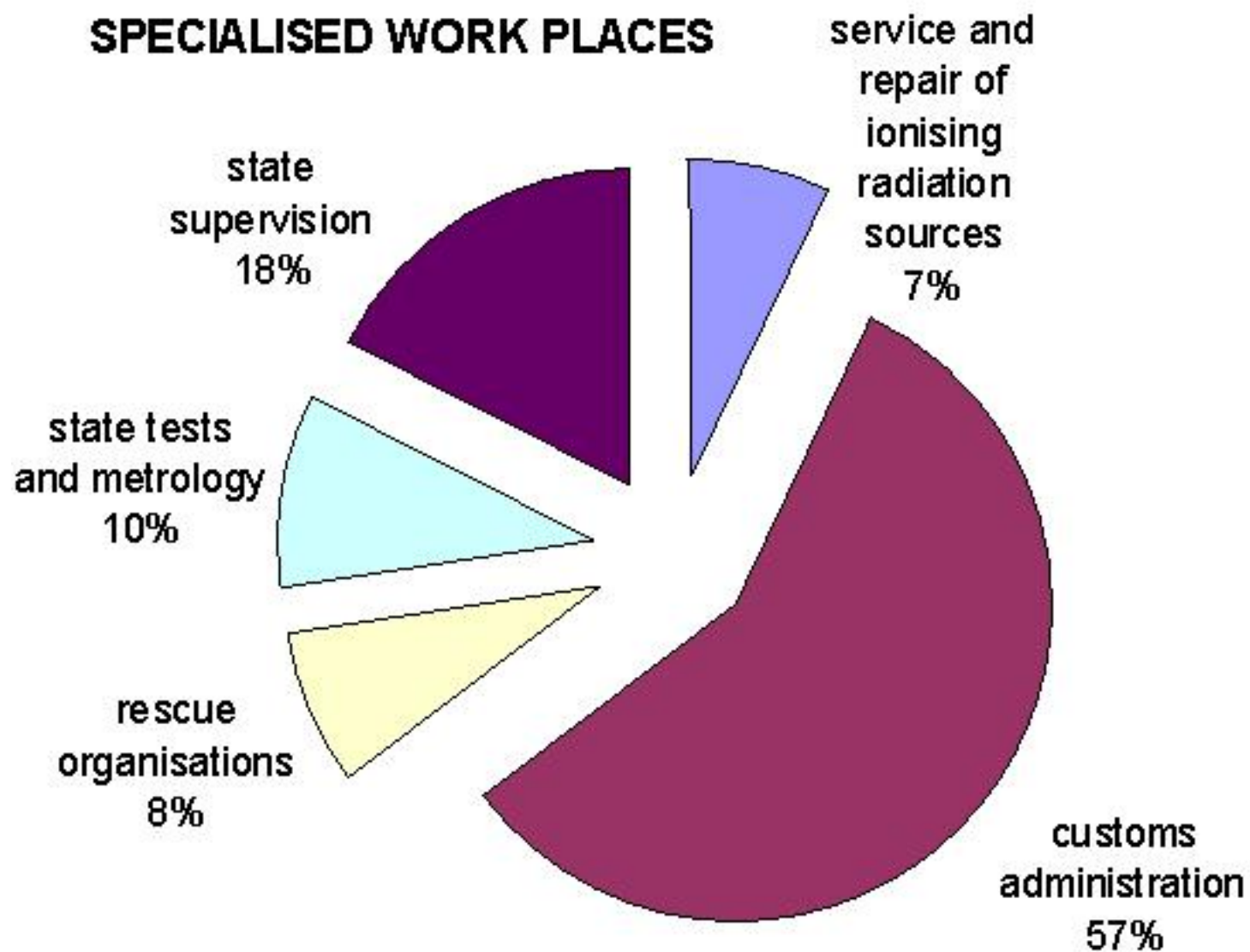
# WORK PLACES



## HEALTH SERVICE



## SPECIALISED WORK PLACES







Type of work		Number of workers (N) / collective annual dose (D) in the range (mSv)											Total number of workers	Average annual dose (mSv)
		0-0.1	0.1-0.2	0.2-0.5	0.5-1	1-2	2-5	5-10	10-15	15-20	20-50	>50		
<b>work sector NATURAL RADIOACTIVITY</b>														
Workplaces with natural radioactivity	N	0	0	2	0	4	0	0	0	0	0	0	6	0,978
	D	0,00	0,00	0,62	0,00	5,24	0,00	0,00	0,00	0,00	0,00	0,00		
<b>work sector SPECIALISED WORK PLACES</b>														
Rescue organisations	N	0	0	0	10	6	0	0	0	0	0	0	16	1,012
	D	0,00	0,00	0,00	8,39	7,80	0,00	0,00	0,00	0,00	0,00	0,00		
Customs officers	N	19	18	46	32	0	0	0	0	0	0	0	115	0,322
	D	1,17	2,53	13,46	19,86	0,00	0,00	0,00	0,00	0,00	0,00	0,00		
Service and repair of sources	N	0	0	3	0	6	0	0	0	0	0	0	9	0,961
	D	0,00	0,00	0,73	0,00	7,93	0,00	0,00	0,00	0,00	0,00	0,00		
Not defined	N	0	9	55	93	67	11	2	1	0	0	0	238	0,964
	D	0,00	1,62	16,85	72,49	83,30	31,40	12,86	10,87	0,00	0,00	0,00		
<b>Total</b>													<b>6740</b>	<b>1,310</b>

# **REGULATORY SUPERVISION OF THE RADIATION PROTECTION AND RELEASE DURING THE DISMANTLING OF THE GERMAN NPP WÜRGASSEN**

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The NPP Würgassen (KWW) is a commercial, single unit boiling water reactor with a capacity of 670 MWe. It was commissioned by PreussenElektra and constructed by AEG/KWU between 1968 and 1971. This NPP was finally shut down in 1994. The decommissioning, started in April 1997, is still under progress. Up to now, approx. 9,500 Mg of various materials (e.g. metal scrap, cable, concrete) have been deconstructed and released from the site.

In the following we will describe the contribution from the authority and the independent expert during the radiological characterisation, the dismantling work, and the material flow up to the release. Special focus will be on the necessary control steps and the documentation regarding the dismantling work and the procedure of release. There is always a close fit between the radiation protection and the release of material on the basis of the radiological characterisation.

## **Radiological characterisation**

First, the NPP management defines a system or a part of a system which shall be taken down. In order to investigate the composition of radioactivity, a so called sampling plan is established by the management. The sampling plan contains all locations where samples (material, scrape or wipe tests) should be taken. The expert then checks the sampling plan. If necessary he adds or changes some locations. In agreement with the expert the NPP operator fixes the exact sampling spots at the system components according to the sampling plan. At this point aspects of radiation protection become important, e.g. the dose rate at the sampling location or the access to the work place have to be considered. Based on this knowledge

further arrangements for radiation protection are made with respect not only to sampling but also to later dismantling.

The NPP management summarises the results within a so called result report after the samples have been taken and analysed. Sample taking and its analysis are spot checked by the independent expert. The most important outcome of the result report is the nuclide vector which is assigned to a system, a spatial area, or parts of it. The nuclide vector describes the composition of radioactivity with respect to the expected disposal objective. On the base of this nuclide vector the result report defines the conditions of the dismantling work, i.e the dismantling methods in combination with the respective radiation protection arrangements. The expert examines this report and creates a report of examination for the authority. If there are disputed points, a detailed discussion between the expert, the NPP management, and the authority can follow before a decision is made.

### **Dismantling of the investigated system**

After the preliminary investigations the NPP operator describes the dismantling unit in a so called working map. The working map establishes all important points of the dismantling unit on a practical level: the detailed procedure of decomposition, the arrangements for radiation protection (e.g. shielding, working tents, protective clothes, etc.), the decontamination techniques, and the measurements intended. Each dismantling unit and therefore each working map only comprises material of one definite nuclide vector in order to separate material according to its different radiological features. Furthermore, it is assured that the activity measurements are carried out with correct calibration.

The working map has to be checked by the independent expert who, if necessary, makes remarks or adds recommendations. In case the authority approves the working map, possibly with additional orders, the dismantling unit can start.

In the process of dismantling most of the material will be cut and put into boxes (skeleton container) of about 600 l volume. For the purpose of definiteness, an identity card is assigned to each box on which all important information about the content can be found, e.g. the origin of the material, the nuclide vector, the decontamination procedures carried out, and the results of measurements. Spot checks by the experts are marked on the identity card. Recently this was replaced by an electronic data processing system. A simplified identity card which a unique barcode is still assigned to the respective box. The modified identity card and the content of the electronic data processing system are further spot checked by the experts.

## **Procedure of release**

In most cases the metal scrap is decontaminated by medium blasting or, less frequently, by high pressure water. If control measurements show a sufficient low contamination the material is brought to the preliminary measurement. This preliminary measurement shall prove that the surface contamination is sufficient homogeneous and in the order of the clearance levels (see below). For other materials, such as cable or concrete, adjacent control steps are established due to the dismantling and decontamination techniques used. These control steps are laid down in a so called release procedure plan and are spot checked by the expert. There is one release procedure plan for each material and disposal objective.

Specific nuclide clearance levels are connected with each disposal objective, e. g. with the unrestricted release of material. That means the mass specific activity (Bq/g) and the surface contamination has not to exceed the specific clearance levels. The NPP management has to prove a contamination below these clearance levels. The final proof is made by a decisive measurement. In most cases this is carried out by a gamma counting measurement of the whole box filled with the respective material. A pre-installed calibration is chosen with respect to the nuclide vector. The results of the measurement are documented. The NPP management creates a complete release documentation containing the information from dismantling and decontamination in combination with the control steps.

In addition to the NPP management, the expert performs spot checks during the measurements, takes samples and analyses them independently. He examines the release documentation and gives a recommendation on the release to the regulatory authority. Finally, the authority checks all the facts and allows the release of the material from regulatory control, if possible.

# MEASUREMENT AND RELEASE OF LOW-LEVEL CONTAMINATED MATERIALS FROM NUCLEAR POWER PLANTS INTO ENVIRONMENT

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## **Abstract**

Release of low-level contaminated solid materials originated from the operation or decommissioning of nuclear power plants into the environment for their recycling, reuse or liquidation is one of the main ways for the minimization of radioactive wastes production. Regulated measurement of contamination and quantitative determination of radionuclides in materials to be released are relatively complicated but manageable processes. AllDeco operates 2 measuring devices for materials release. For this purpose measuring chambers of RTM type are used for the determination of total gamma activity of radionuclides in materials. Activities of alpha and beta radionuclides are calculated by statistical results processing based on radionuclide vectors established in the laboratory for samples of similar materials. Possible spatial inhomogeneities of the contamination in measured materials can be identified by the devices software.

The measurement of total gamma activity is completed by controlling spectrometric determination of present gamma radionuclides using NaI or Ge detectors.

Limits for the release are measured and observed according to assigned accuracy for measuring chambers as authorized official measures. The stipulated reference levels are controlled by selective measurements of the surface alpha and beta activities and by the determination of the ratio of main gamma radionuclides.

## Introduction

Important aims of an effective operation or decommissioning of NPP are recycling and reuse of materials and equipment.

The company AllDeco prepares and realizes the clearance measurement and free release of material into the environment by means of the measurement of its mass activity by a clearance measurement station RTM 661/540*Inc*.

The measurements are realized according to the legislative standards in the Slovak Republic [1, 2] and in the Czech Republic [3, 4] and according to the national metrological standards.

The company AllDeco performs free release measurement using one station RTM 661/540*Inc* at the NPP A-1 (Jaslovské Bohunice, Slovak Republic) at decommissioning works and the second, mobile station at the NPP EBO (Jaslovské Bohunice, Slovak Republic) and NPP Temelín (Czech Republic) during operation of these facilities.

The measurement station RTM 661/540*Inc* is an accepted device for measurement of the mass gamma activity of the materials released from NPP according to DIN standard No. 25 457 in Germany.

## Measurement of mass activity of materials

Measurement of mass activity of materials for release into the environment is based on the accurate determination of the activity and the mass activity of gamma radionuclides in materials. The contents of other present radionuclides (alpha, beta) are calculated by a radionuclide vector (RNV).

In decision making about performance/non-performance of limit conditions of radionuclide contents in measured materials it is needed adhere to the term:

$$\sum_{i=1}^n \frac{a_{mi}}{a_{mLi}} \leq 1$$

where  $a_{mi}$  ... mass activity of radionuclide  $i$  [Bq.kg<sup>-1</sup>] in material  
 $a_{mLi}$  ... release limit for radionuclide  $i$  [Bq.kg<sup>-1</sup>] from Law or Regulation  
 $n$  ... number of radionuclides in material

The mass activity limits in the Slovak Republic [1] vary from 300 Bq.kg<sup>-1</sup> (Mn-54, Co-60, Cs-137, Am-241, Pu-239, ...) up to 3 000 kBq.kg<sup>-1</sup> (Ni-63, ...).

In practice, the measured material is non-homogeneously contaminated. Therefore, the measurement is averaged for the mass of 300 kg (in 200 dm<sup>3</sup> drum) and also the homogeneity of the activity distribution in the measured volume is controlled.

## **Results of clearance measurement and material release from the NPP's A-1, V-1, V-2 in Jaslovske Bohunice and NPP Temelin**

### **1. Applied procedure**

For the clearance measurement and free release of materials the following procedure is applied:

- a) Definition of the batch, types and size of wastes
- b) Sampling of wastes
  - Radiochemical determination of activities of key nuclides in taken samples
  - Statistical processing of results obtained from radiochemical analysis
  - Determination of RNV
- c) Measurement of mass activity of material by station RTM 661/540 includes:
  - Geometry calibration of measured materials
  - Measurement of the mass activity of materials in drums
  - Control spectroscopy measurement of contents and ratio of gamma nuclides in the measured material
  - Control surface contamination measurement (control of reference levels) on some parts of the material
  - Issue of the protocol and labelling of measured drums, bags and palettes
  - Control sampling from measured materials, radiochemical analysis and verification of the RNV
- d) Dosimetry and government control of measured materials
  - Control of protocols and documentation



- e) Material release from the controlled area in sealed drums, containers, ...

## 2. Preparation for the measurement - examples

The schema of a fuel container T-12 and the identification of sampling places are shown on the Fig. 1. These fragmented and decontaminated fuel containers are measured and released from the NPP's in Jaslovské Bohunice. The electrochemical sampling from the surface of the fuel container T-12 for the radiochemical analysis is shown on the Fig. 2.

Determinations of the surface activities were then statistically processed as it is shown in the Tab. 1. Files with experimental values of the surface activities from containers were tested for different statistical distributions. The method of quantile regression was included in calculations. This method was used when the course of mean value was not interesting but the course of extreme value was defined by a conservatively selected quantile. Observed low level surface activities had a log-normal statistical distribution as resulted from statistical testing of collected data.

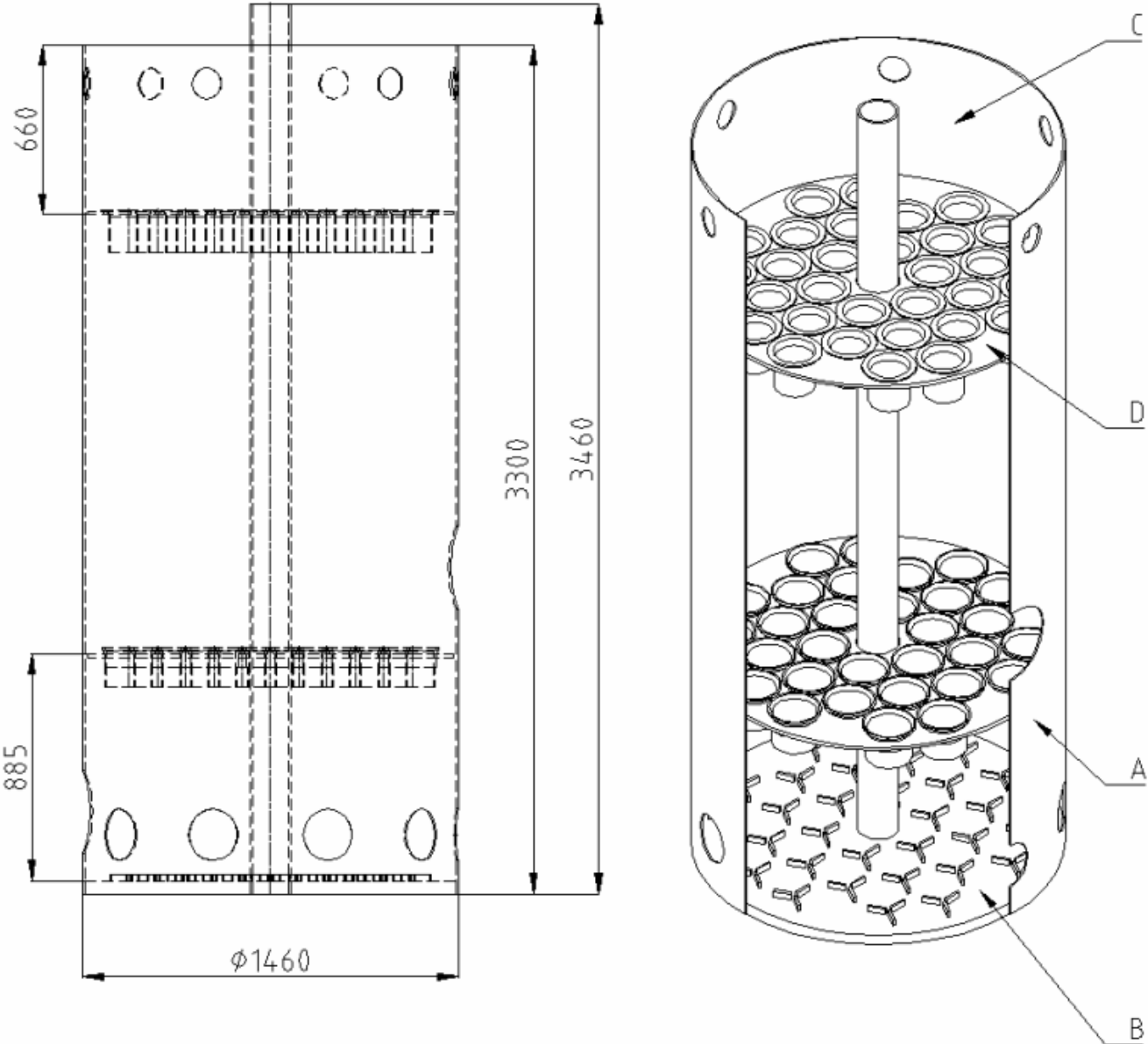
**Tab. 1: Surface activities [Bq.cm<sup>-2</sup>] and statistical data calculated at the significance level 95 % ( $\alpha = 0.05$ ) for samples from fuel containers T-12**

Set No.	Container	Cs-137	Co-60	Pu-239, 240	Pu-238	Am-241	Cm-244	Sr-89, 90
1.	125A/1-10	7.038	7.299	0.019	0.020	0.029	0.003	0.369
2.	125B/1-10	15.140	33.146	0.008	0.015	0.033	0.016	0.643
3.	125C/1-10	7.124	15.670	0.005	0.005	0.028	0.003	0.118
4.	125D/1-10	24.070	31.417	0.028	0.026	0.048	0.007	1.385
5.	293A/1-10	4.678	1.487	0.002	0.002	0.005	0.002	0.936
6.	293B/1-10	4.557	6.149	0.002	0.002	0.007	0.003	0.172
7.	293C/1-10	4.108	1.293	0.002	0.005	0.020	0.002	7.226
8.	293D/1-10	17.369	36.159	0.075	0.070	0.137	0.028	5.426
9.	456D/1-10	19.976	24.494	0.001	0.001	0.005	0.002	0.012
10.	456C/1-10	6.733	27.876	0.001	0.002	0.006	0.002	0.006
11.	557C/1-10	2.631	39.787	0.002	0.007	0.007	0.002	0.003
12.	557B/1-10	3.354	61.687	0.001	0.002	0.007	0.003	0.031
13.	557A/1-10	3.118	0.657	0.002	0.002	0.004	0.008	0.023
14.	125A/1-5/3	8.322	0.931	0.008	0.009	0.011	0.001	0.128
15.	125A/1-5/1	11.734	11.462	0.014	0.013	0.014	0.002	0.180
16.	125D/1-5/1	16.250	21.352	0.019	0.020	0.029	0.003	1.166
17.	125D/1-5/3	6.690	1.262	0.003	0.007	0.007	0.002	0.029
18.	538A/1-5/1	17.052	1.100	0.003	0.002	0.034	0.001	0.022
19.	538A/1-5/2	8.140	0.678	0.003	0.050	0.072	0.002	0.017
20.	456B/1-10	4.656	42.334	0.002	0.003	0.007	0.003	0.013
21.	456A/1-10	5.810	3.248	0.002	0.002	0.008	0.005	0.024

<b>median</b>	7.038	11.462	0.003	0.005	0.011	0.003	0.118
<b>upper limit</b>	1.459	-4.568	0.000	-0.002	0.000	0.002	-0.212
<b>lower limit</b>	12.617	27.492	0.006	0.012	0.023	0.003	0.448

<b>average</b>	9.455	17.595	0.010	0.013	0.025	0.005	0.854
<b>upper limit</b>	6.588	9.457	0.002	0.005	0.011	0.002	-0.005
<b>lower limit</b>	12.322	25.733	0.017	0.021	0.039	0.008	1.712

**Fig. 1: Scheme of the fuel container T-12 and the identification of places for the electrochemical sampling of the surface activity**



**Fig. 2: The electrochemical sampling from surface of the fuel container T-12 for radiochemical analysis of taken samples**



On the basis of radiochemical analyses and after statistical results processing, the radionuclide vectors were defined for different batches of materials. The results of radiochemical analysis of some samples from NPP's V-1, V-2 (Jaslovske Bohunice) and NPP Temelin and percentage representation of radionuclides for the RNV determination are shown in Tab. 2 and Tab. 3.

**Tab. 2: Radionuclide vectors for material batches from NPP's V-1 and V-2 (Jaslovske Bohunice)**

	<b>Batch No 1</b> <b>Paper, Spoil, Steel</b>	<b>Batch No 2</b> <b>Filters</b>	<b>Batch No 3</b> <b>Wood</b>	<b>Batch No 4</b> <b>Textile</b>
<b>Gamma radionuclides</b>	<b>RNV 1</b> <b>(% repr. in mixture)</b>	<b>RNV 2</b> <b>(% repr. in mixture)</b>	<b>RNV 3</b> <b>(% repr. in mixture)</b>	<b>RNV 4</b> <b>(% repr. in mixture)</b>
Co-60	19.526	0.466	47.596	4.780
Co-58			2.555	9.215
Cs-137	19.462	0.245	25.047	1.629
Cs-134			1.560	2.740
Mn-54	7.540	0.221	2.012	5.283
Sb-124			2.629	9.145
Sb-125				2.000
Ag-110M	2.534	0.062	3.622	58.396
Fe-59				1.353
Zr-95				0.554
Ru-103				0.282
Ce-144				3.295
<b>Other controlled radionuclides</b>				
C-14	31.223	98.626	8.017	0.264
Ca-41	0.452	0.011	0.120	0.004
Ni-63	8.379	0.123	2.226	0.156
Sr-90	5.506	0.141	1.404	0.371
Nb-94	0.811	0.018	0.703	0.121
Tc-99	2.692	0.058	2.055	0.388
I-129	0.624	0.009	0.135	0.004
Pu-238	0.161	0.005	0.043	0.006
Pu-239, 240	0.585	0.009	0.140	0.008
Am-241	0.505	0.007	0.136	0.004

**Tab. 3: Results of radiochemical determinations in waste batches S1 and S2 from NPP Temelin and percentage representation of radionuclides for RNV determination**

Radionuclide	Batch S1					Batch S2					
	Sample No.; $a_m$ [Bq.kg <sup>-1</sup> ]					Sample No.; $a_m$ [Bq.kg <sup>-1</sup> ]					
	1	2	3	Median	% repr.	1	2	3	4	Median	% repr.
<b>C-14</b>	17.000	21.000	26.000	21.000	11.221	17.000	21.000	28.000	13.000	19.000	22.403
<b>Ca-41</b>	0.300	0.400		0.350	0.187	0.300	0.400			0.350	0.413
<b>Ni-59</b>	0.600	0.400	0.450	0.450	0.240	0.400	0.500	0.580	0.370	0.450	0.531
<b>Ni-63</b>	6.400	7.200	5.100	6.400	3.420	9.100	8.200	7.300	4.500	7.750	9.138
<b>Sr-90</b>	10.100	4.900	2.900	4.900	2.618	8.500	6.800	5.000	3.900	5.900	6.957
<b>Nb-94</b>	0.700	0.900	0.360	0.700	0.374	0.700	0.400	0.510	0.390	0.455	0.536
<b>Tc-99</b>	9.900	9.100	8.700	9.100	4.862	12.400	8.800	8.900	5.100	8.850	10.435
<b>I-129</b>	0.700	0.700	0.720	0.700	0.374	0.700	0.600	0.790	0.290	0.650	0.766
<b>Cs-137</b>	0.380	0.920	0.480	0.480	0.256	0.470	3.120	0.670	0.220	0.570	0.672
<b>Pu-239, 240</b>	0.200	0.200	0.210	0.200	0.107	0.100	0.200	0.250	0.110	0.155	0.183
<b>Am-241</b>	0.200	0.200	0.230	0.200	0.107	0.100	0.200	0.310	0.160	0.180	0.212
<b>Mn-54</b>	2.340	249.000	101.000	101.000	53.967	0.930	2.170	107.000	21.300	11.735	13.837
<b>Co-58</b>	1.330	9.890	16.700	9.890	5.285	0.450			28.000	14.225	16.773
<b>Co-60</b>	0.820	5.780	27.700	5.780	3.088	0.220	1.520	26.000	2.460	1.990	2.346
<b>Zn-65</b>			0.660	0.660	0.353			1.040		1.040	1.226
<b>Nb-95</b>	2.290	5.740	24.300	5.740	3.067	0.900	0.560	36.700	0.810	0.855	1.008
<b>Zr-93</b>	1.200	3.290	10.600	3.290	1.758	0.560		16.300	0.680	0.680	0.802
<b>Sb-124</b>	1.230	24.200	15.800	15.800	8.442			17.700	1.090	9.395	11.078
<b>Sn-126</b>			0.510	0.510	0.273			0.430	0.730	0.580	0.684
<b>Sum</b>				<b>187.150</b>						<b>84.810</b>	

### 3. Mass activity measurement

The measurement of the mass activity of material was realised by the station RTM 661/540Inc (RADOS). The measurement chamber is surrounded by detectors on all six sides. Here, a geometrical factor (chamber surface area divided by the effect detector area) of 60% has been achieved. An external screen incorporating special measures such as the cladding of the holder profile with rod steel ensures a good background reduction, giving in turn short measurement times and excellent detection sensitivities (3 Bq.kg<sup>-1</sup> for Co-60, t = 60 s in total mass of 100 kg in the drum).

For the geometry calibration of measured materials a calibration dummy for a 200 dm<sup>3</sup> drum is used, which consists of a frame carrier that can carry 2 mm thick steel plates. These plates have to be selected in number and arrangement inside the frame to correspond with the geometrical and mass conditions of the geometry to be calibrated (Fig. 3).

**Fig. 3: Calibration measurements with various filling levels and densities of measured materials for 200 dm<sup>3</sup> drums**



The control measurement of contents and ratios of gamma nuclides (verifying of the RNV) is realised by the gamma spectrometric system placed in the left rear corner of the RTM station. At clearance measurement in the NPP A-1 (SE-VYZ a. s.), gamma spectrometric system with NaI (Tl) scintillation detector (CANBERRA) is used and in the NPP's V-1, V-2 and NPP Temelin gamma spectrometric system with HPGe detector (ORTEC) is used.

The control surface contamination measurement (control of reference levels) is realised by a direct measurement of the surface alpha- and beta-activity by the alpha/beta spectrometric system (SARAD) with the Si-detector (PIPS detector, CANBERRA).

After the measurement of the mass activity of the material, the protocol is printed and measured drums, bags or palettes are labelled. The control sampling and following radiochemical determination for improvement of the RNV are performed from measured materials.

#### **4. Dosimetry and government control of measured materials**

Measured material stated as clear for the free release is verified by the dosimetry of the NPP by random control measurement of the surface contamination. After this, the material with confirmed protocols is released from the controlled area of the NPP in sealed drums or containers.

Government (authority) control includes:

- Control of protocols and documentation
- Issue of license for clearance measurement and material release
- Control of metrological verification of devices (scales, detectors, ...)
- Control of the license for measurement

## 5. Results of clearance measurement

After fragmentation, electrochemical and ultrasonic decontamination, 30 tons of stainless steel fragments from fuel containers T-12 were measured by the RTM station at the NPP A-1 in years 2002 – 2003. Approximately 15 tons of material were released into the environment according free release limits ( $300 \text{ Bq.kg}^{-1}$ ). The rest of materials (approximately 15 tons) was slightly activated (by Co-60) up to levels  $500 - 1\,500 \text{ Bq.kg}^{-1}$ . This rest of steel is stored at the NPP site for 10 – 15 years.

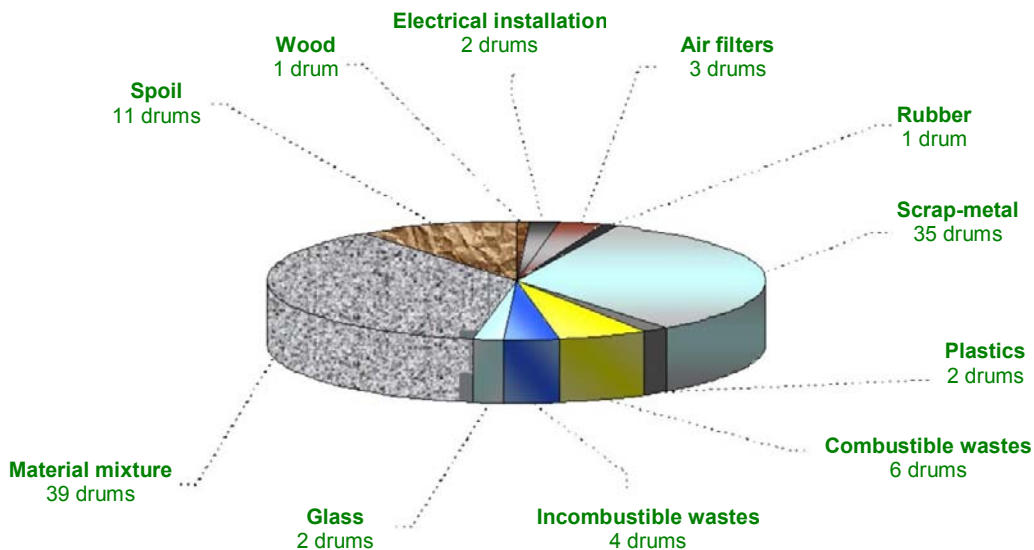
At the NPP V-1, 2 586 kg of materials from the standard operation were measured (spoil, paper, filters, active carbon, textile and wood) in the year 2003. About 1 414 kg of materials were released into the environment following free release limits. The clearance measurement at the NPP V-2 is realised in this time.

The clearance measurement and material release at the NPP Temelin were realised by the RTM 661/540 station in November 2002 (Fig. 4). Material balance of measured waste drums at the NPP Temelin is shown in the Fig. 5. The NPP Temelin is in operation for 3 years.

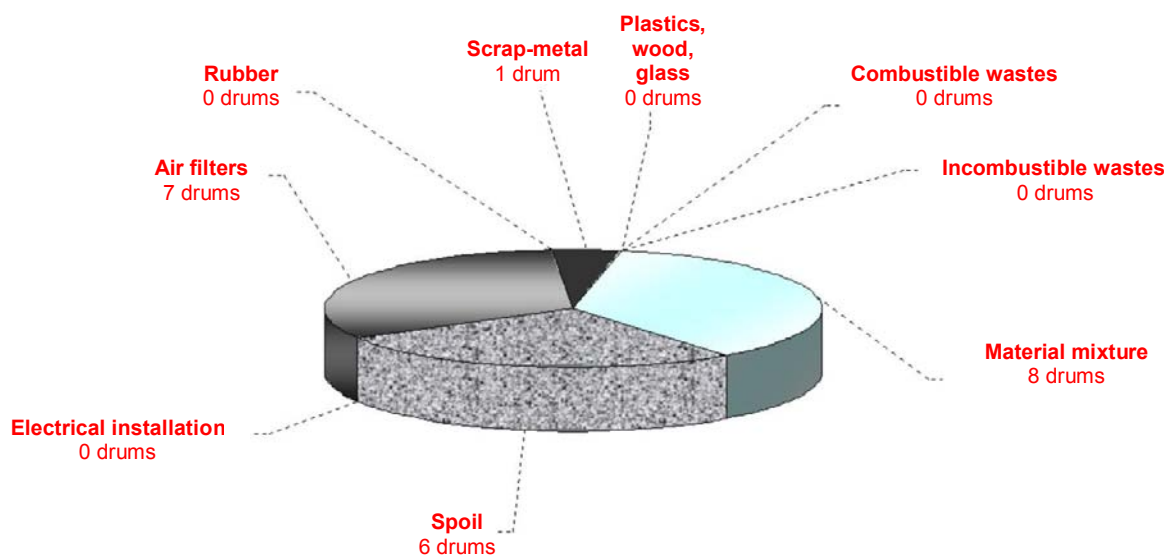
**Fig. 4: Measurement of wastes in NPP Temelin in November 2002**



**Material balance of drums suitable for the free release into the environment (106 drums, 83 %)**



**Material balance of drums unsuitable for the free release into the environment (22 drums, 17 %)**



**Fig. 5: Material balance of the measured waste drums for the free release into the environment from measurement by the RTM 661/540 station in the NPP Temelin in November 2002**



## **Conclusion**

Measurement and free release of materials from the operation or decommissioning of the NPP's in the Slovak Republic and the Czech Republic are developed up to the practice.

The Company AllDeco started this process in the practical manner in the year 2002. The operation of two measurement stations is simple and safe according to all regulatory and authority requirements valid in this area. Free release limits and measurements procedures are similar and comparable to those ones in other countries in EU.

## **References**

1. Act of the Slovak National Council No. 272/1994 Coll., on the Protection of Human Health
2. Decree No. 12/2001 Coll., on the Requirements on Radiological Protection Assurance
3. Law No. 18/1997 Coll., on Peaceful Utilisation of Nuclear Energy and Ionising Radiation and on Amendments and Additions to Related Acts
4. Regulation No. 307/2002 Coll., of the State Office for Nuclear Safety on Radiation Protection

# RADIONUCLIDE DISPERSION CALCULATION IN ENVIRONMENTAL RADIATION MONITORING SYSTEM OF THE PAKS NPP

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## Introduction

The new Environmental Radiation Monitoring System of the Paks NPP in Hungary consists of three radiation release measurement posts (placed into the two ventilation stacks of the four units and into the ventilation stack of the spent fuel intermediate storage building), 9 radiation monitoring stations and 11 gamma-radiation measurement posts placed more or less evenly around the plant.

During normal operation, the system monitors and integrates the amount of the actual releases for the authorities. In case of accidental releases of radioactivity the system

measures the releases in the stacks;

using *real-time simulation model* and meteorological data estimates the resulting gamma-radiation field at the positions of the monitoring stations and posts; compares the calculated dose values with the measured dose rates integrated for the time period;

if the measured values are significantly higher, that means the containment is leaking and this leakage is estimated;

adding up the leakage to the measured releases the model calculates the actual gamma dose caused by the cloud and by deposition on the soil surface, and predicts those for all measurement posts and for 21 villages within a 30-km radius around the plant.

Besides monitoring the normal operation of the plant the purpose of the system is to determine the environmental dose burden caused by atmospheric dispersion of the radionuclides during accidental releases as well; the system plays an active role in the mitigation of the consequences. This means that the system is classified similarly to the other safety-related equipment of the plant.

## The measurements

There are three possible sources of regular radioactive release: the two ventilation stacks of the Nuclear Power Plant (NPP) and the ventilation stack of the Spent Fuel Intermediate Storage (SFIS). These are surrounded by 9 complex measurement stations type A and 11 gamma dose-rate measurement posts type G.

The stations and measurement points are not located evenly. We had to take into account the location of existing roads, buildings, private properties, power lines, etc. The detection range is shown by solid lines across the body of the stations type A and gamma dose-rate measurement posts type G on the Fig. 1. The width of these ranges are calculated to indicate at least 10% of the maximal

centerline activity concentration of the actual plume released in case of neutral dispersion conditions. The average radius of the measurement posts is 1.4 km.

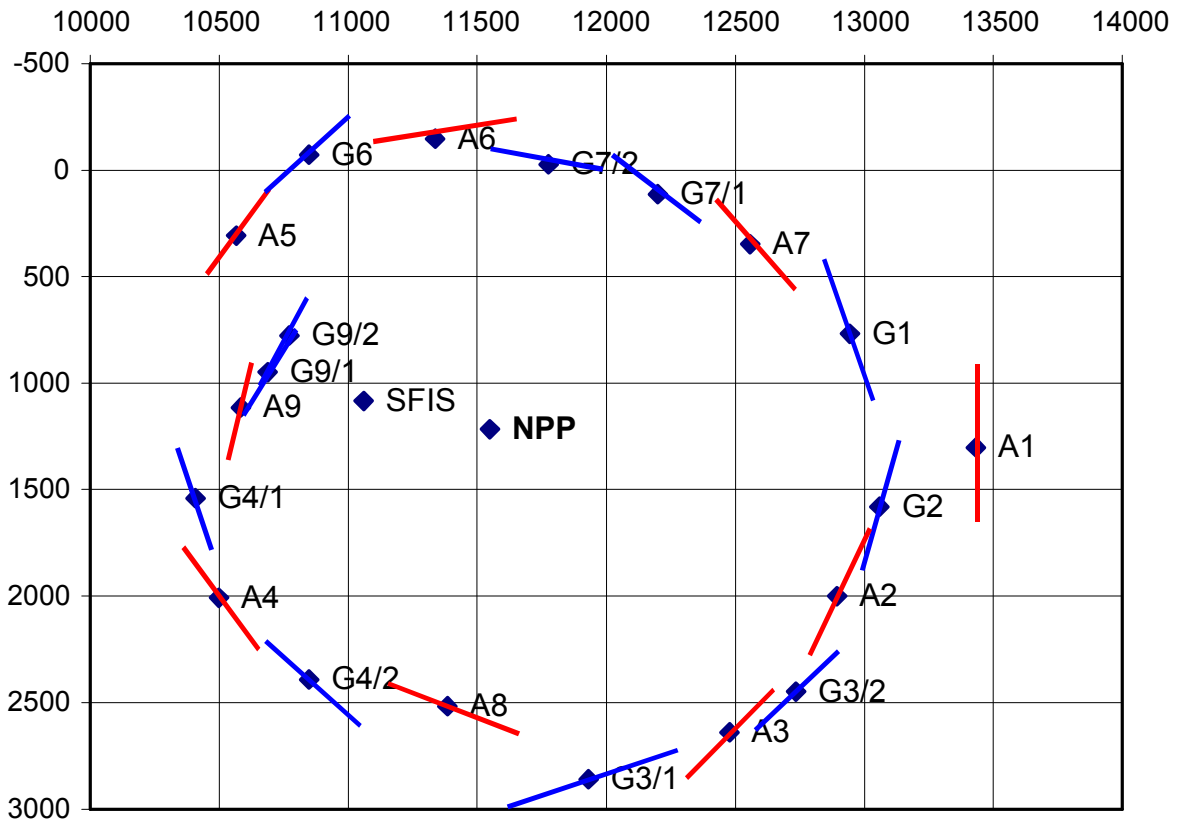


Fig 1. Stacks of the Nuclear power plant (NPP) and Spent fuel intermediate storage (SFIS) surrounded by measuring stations (A – red marked) and detectors (G – blue marked). The sensitivity areas are shown by lines, the grid dimension is given in m.

The actual gamma dose radiation level resulted by normal operation of the nuclear power plant is less than 0.1% of the natural radiation level (even the 10 % variation of the natural level is higher by two orders of magnitude); this is therefore immeasurable. The G-type measurement posts are useful in accidental situations only. They consist of a BITT-made gamma dose-rate detector; a solar cell with control electronics and a rechargeable battery; a radio modem to transmit the measured data. Having no connecting cables and external power sources it is easy to make them strong enough to withstand the maximal predictable earthquake.

The A-type stations are equipped with the same type of gamma dose-rate detector. These stations are situated in aluminum containers used by the railroad freight traffic. Along with the gamma dose-rate measurement the following time-integral measurements are carried out continuously by these stations:

- aerosol gross beta-activity on-line measurement;

elementary iodine-activity concentration on-line measurement;  
organic iodine-activity concentration on-line measurement;  
additionally a high-flow-rate sampling unit for low-concentration off-line (laboratory) measurements is applied.

The measurements of the released activity can be divided to on-line and off-line ones. Only on-line measurement values - transferred to the central data processors immediately - are used for dispersion calculations. These are: the PING (Particulate, Iodine and Noble Gas) monitor produced by MGP Instruments; the NEKISE (Nuclide-specific Noble Gas Analyzer) produced by the Budapest Technical University and a wide range gamma dose-rate detector made by BITT. The latter is needed to measure the high radiation levels during accidental situation caused by occasional failure of the forced ventilation.

### **Data acquisition and display**

The primary goal is to collect all the data measured in the stack, in the spent fuel storage and in the fields as well as data from the meteorological tower and present them in the control room and in the radiation protected control post. (The central data processors get the meteorological data - wind velocities and directions, air temperatures, precipitation, etc. measured at different elevations of the 120 m high meteorological tower using radio links.) Together with the measured data the environmental dose fields calculated by the dispersion model are also displayed.

### **The dispersion model**

Besides the presentation of the measured data, the system calculates the dispersion of the radionuclides in the 30 km vicinity of the NPP in normal and accidental situations, too.

The dispersion of the radionuclides released by the NPP is a sophisticated phenomenon. There are four parallel processes to be modeled continuously in time and space:

- the release of radioactive gases and aerosols: the volume of the outflow from the stack, and the initial isotope concentration of it is measured continuously;
- to calculate the dispersion of the released radioactive noble gases, aerosols and elementary iodine we have to take into account the actual meteorological conditions (e.g. wind direction, wind speed and turbulence as function of space and time);
- as time passes the isotope concentration of the radioactive materials changes continuously due the chains of radio-active decay; and dry (and wet) deposition;
- the radiation in any point on the ground level around the NPP must be calculated as an integral of distance taken over the (ever changing) air isotope concentration in the vicinity of that point, as well as the radiation of the fall-out accumulated around on the soil surface so far.

All these phenomena are calculated using the three-dimensional Gaussian puff models. The basic ideas are shown in [1] and [2], some details concerning calculations in [3]. According this method the process of continuous release is approximated by the finite sequence of so called 'puffs', each corresponding to a 10-minute release. Each puff contains a given amount of different radioactive materials, determined by the intensity of the release and of the actual time period of the sequence after the release.

The central point of the puffs is moving according to the actual wind vectors and the dilution of the puffs is determined by the actual air turbulence, described in our case by the discrete Pasquill stability classes. The continuous drifting and dispersion are approximated by the sequence of discrete steps. The time base is 10 minutes, therefore the model releases a new puff in each 10 minutes.

Due to the reflection from ground surface and to the temperature inversions the vertical Gaussian distribution of the puffs may be modified. The height of the barrier layer of the inversion at each Pasquill stability class is taken from the literature.

We take into account plume depletion procedures due to dry and wet deposition. We follow the radioactive decay as well. The time period elapsed between the reactor incident or accident and the release to the environment, the time of the dispersion of the puffs and the decay time of radionuclides after the fall-out to the soil are all taken into account. Calculating the dose effects of these decays all the most important daughter isotopes are taken into account, too. The calculation is made for 58 different isotopes (the different forms of iodine isotopes are calculated separately). The conversion factors for the internal radiation doses - by inhalation and due the food chain - are selected according to the international recommendations. The food chain depends upon the actual season of the release - therefore we cannot omit the seasonal effects.

### **Normal mode calculations**

During normal operation, the on-line measurement posts in the fields are unable to detect excess radiation; therefore the calculated gamma dose-rate distribution caused by noble gas releases is presented on the display of the information system, together with the 10 minute average of the wind direction and velocity. The calculations performed are based on the activity measurements in the two stacks of the NPP.

### **Accident mode calculation**

#### *Source term estimations*

We use the following algorithm to determine the places and the equivalent elevations of the releases:

1. We measure the releases made through the ventilation stacks of the NPP and the SFIS. These are direct measurements which are accepted unconditionally.
2. Using the measured releases in the stacks and the measured actual meteorological conditions we simulate the dispersion process and calculate the expected gamma dose at the A-type stations and G-type posts.
3. If the radiation levels measured by the A-type stations and G-type posts are significantly higher than the calculated values, we take the decision that the containment is leaking somewhere and there are releases bypassing the stacks.
4. The excess amount of the activity is re-calculated back to the release point assuming the height of the release as 25 m (half of the height of the containment).

In steps 1 and 2 we perform a simplified calculation, determining only the gamma dose caused by the puffs for 20 points (9 A-type and 11 G-type measurement points). We are taking into account the fall-outs accumulated on the soil surface so far, too.

Now, after step 4, using the new source terms (stack and/or containment) we re-calculate the dispersion process - this time not only for the A and G type measurement posts but in the whole vicinity of the power plant; and not only the external gamma doses but the doses caused by inhalation and via food-chain are also calculated.

It is very important to have a good estimate of the initial isotope composition for calculating the environmental dose burden from leakage via the reactor building. Basically there are four types of isotope composition:

Successful LOCA (Loss of coolant accident) - only some fuel rod claddings are ruptured;

Unsuccessful LOCA - the core is melted but the debris remains inside the reactor vessel;

Unsuccessful LOCA - the debris of the melted core penetrates the reactor vessel;

PRISE - water from the primary circuit penetrates to the secondary circuit and the release comes from there

The first three cases start with the gap activity (gap on the cladding of the fuel rod) and the fourth with water activity. These are the two initial isotope compositions for the very beginning of the calculations.

### *Dose predictions*

The dose for different nodal (grid) points inside the vicinity of the plant can be calculated inserting

- a) the source term estimated,
- b) the isotope composition accepted and
- c) the meteorological data measured

as inputs for the dispersion model described above. As far as the source term cannot be predicted for a longer period of time, we cannot make reliable estimations for longer period than 2-6 hours even if we assume that the meteorological and release conditions do not change. There are two types of doses calculated to all nodal (grid) points:

Effective gamma dose increment from external radiation. This value is calculated adding up the radiation coming from the clouds and radiation coming from the fall-out on the soil.

Inhaled iodine radiation dose increment of the thyroid glands of the children.

### **Computational constraints**

The calculation is made in the following steps:

#### 1. Keeping track of the puffs:

in each time step of 10 minutes a new puff is born;

the actual parameters of this puff (size, isotope content, location etc.) are calculated and stored, as long they are inside of the 30 km-radius circle of interest;

the puff is deleted from the data storage if it leaves the territory, or its radiation decays significantly. A cut-off at 144 puffs (one day release) is also applied for the total number of puffs within the territory, so that the puffs can not accumulate by thousands even in case of no wind at all.

#### 2. Calculating the radiation

According to the aging of each puff, the isotope content and the intensity of radiation is changing, too. Calculating the dose rate for a given point, located inside the territory, the impact of

each puff has to be taken into account, adding up their radiation weighted with the actual distance. The same is true for the fall-outs, accumulated on the surface so far.

All these values are calculated for the following points:

Polar coordinates (between 0 and 10 km radius- 1 km resolution, between 10 and 30 km radius - 2 km resolution; angular resolution is evenly 5 degrees), that makes 20 circles x 72 directions = 1440 points;

9 A-type stations, and 11 G-type stations;

11 pre-selected measurement points (K-type check points for mobile laboratory measurements);

point of the maximal radiation inside the 3 km radius around the plant;

21 villages within the 30 km radius around the plant.

That makes close to 1500 points to calculate and integrate. All these values are put into the archives as well.

## Conclusions

The basic goal of the Environmental Radiation Monitoring System is to provide complex and reliable information about the releases in all operating modes to facilitate the adequate estimation of the situation and to promote the decision making.

Thanks to the astonishing development in the digital technology and to the state-of-the-art, up-to-date measurement techniques, a new level of confidence can be reached. Unpredictable radioactive leakage of the containment can be detected and the radiological situation of a relatively large area can be calculated and predicted. A very reliable system can be constructed withstanding earthquake and protected against single failure.

Based on reliable and detailed measurement data, advanced simulation methodology and well-designed graphical user interface, an easy-to-use operator advisory system can be created to help the decision making in the very first and most difficult period of a nuclear accident.

It is very important that the same system is used with the same features during the normal operation of the nuclear power plant, too; this means that the operators are able to get the necessary 'hands-on' training in order to be able to use the system during extreme stress and very unusual situations, too. Shaping the system in close cooperation with plant engineers and operators is indispensable in order to achieve the aforementioned goals.

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# RADIOACTIVE DISCHARGES FROM BOHUNICE NPP

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## Introduction

Bohunice NPP operates 4 PWR reactors VVER type. First two reactors are well known under the name Bohunice V1. They were put in operation in 1978 (Unit 1) and in 1980 (Unit 2). The other two are younger ones and they have been in operation since 1984 (Unit 3) and 1985 (Unit 4). Unit 3 and 4 are named as Bohunice V2.

The construction of both V1 and V2 NPPs is similar but not identical. The original Russian design of the NPPs came from seventies years of 20<sup>th</sup> century. It corresponded to the safety requirements as well as technical abilities of that time.

A process of safety improvement and operation reliability has started immediately after units' commissioning. For example more than 1300 modifications have been implemented at V1 NPP before 1993 followed later by large reconstruction works (1993 - 2000). Based on the results of international review missions carried out at V1 after the reconstruction is obvious that V1 NPP met all the requirements defined by Slovak Nuclear Authority and by IAEA. **Finalising the reconstruction works V1 NPP reached the internationally acceptable level of nuclear safety.**

The similar situation in the trend of safety improvement can be seen also at V2 NPP when the modernization process has started two years ago.

The original radiation protection systems at V1 and V2 are very comprehensive and the important measurements are doubled or even tripled (200% redundancy). Large work in modernisation of radiation protection equipment had been performed in all areas. Modernisation and reconstruction touched the personal dosimetry, contamination monitors at the exit from radiation controlled areas and NPP personal and vehicles gates, accident monitoring systems, environmental radiation monitoring systems, management software and discharge monitoring systems spectrometry laboratories including.

## Gas discharge monitoring system

The aim of the monitoring is the precise determination of the discharged radioactivity, which is the base for the assessment of the public exposure (calculation of the exposure of the member of the critical group. This goal must be achieved during normal operation as well as during accidents. The requirements for the monitoring systems are serious as the activities during the accident could differ to  $10^{10}$  or more and the monitoring conditions could be severe.

## Review of the monitoring

The basic review of the monitoring systems and their measuring ranges for normal operation and accidents is provided in the table below. Both sampling and continual measurements are very important for each event – accident and normal operation. Laboratory measurement of the samples provides more precise results for the assessment of the NPP influence into its surroundings meanwhile the continual monitoring serves to fast response



when any anomaly occurs. Of course the results from the continual measurements also serve for fast estimation of discharged activity especially in case of accident.

	<i>noble gases</i>	<i>particulates</i>	<i>J131</i>
<i>measurement of samples in laboratory</i>	<p style="text-align: center;"><b>gamma spectroscopy</b> (HPGe 60%) sampling is done by the basic monitoring system</p>		
range of measurement	from 100 Bq/m <sup>3</sup> discontinuous samp.	from 1E-4 Bq/m <sup>3</sup> continual sampling	
<i>on-line measurement</i>	<p style="text-align: center;"><b>basic monitoring system</b> (with isokinetic sampling) <b>accident monitoring system</b></p>		
range of measurement	2E3-3E15 Bq/m <sup>3</sup> (Xe133)	1-1E11 Bq/m <sup>3</sup> (Cs137)	2-1E11Bq/m <sup>3</sup>

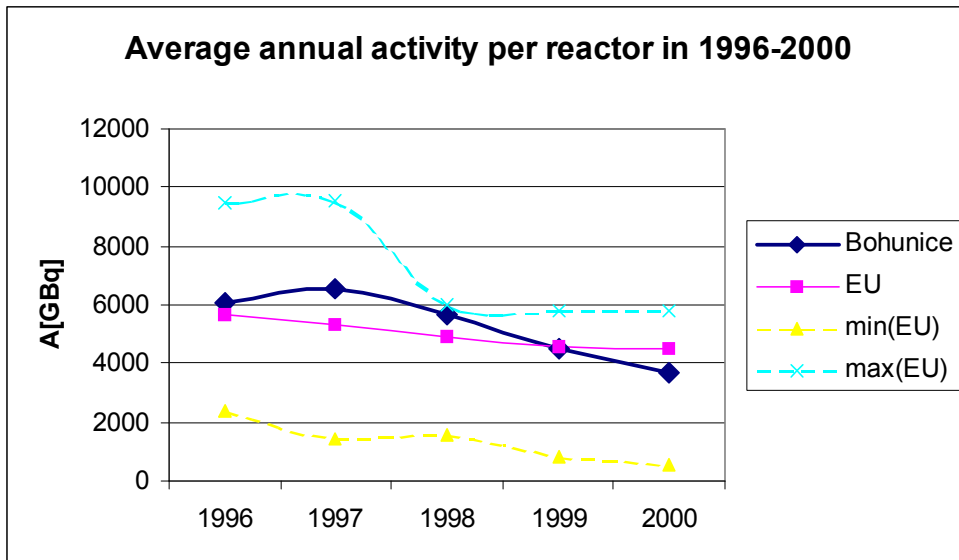
The other radionuclides that are monitored in gas discharges at Bohunice NPP are <sup>3</sup>H, <sup>14</sup>C, <sup>89,90</sup>Sr, <sup>238</sup>Pu, <sup>239+240</sup>Pu and <sup>241</sup>Am. Tritium and carbon are collected in the sorbent and then analysed in the laboratory using liquid scintillators. Strontium and alpha radionuclides are sampled and certified contractors perform their analyses.

Bohunice NPP operates two modern continual noble gas discharge spectrometry systems (MVP). High sensitivity of the measurements is ensured by pressurized flow (700 kPa) through the monitor.

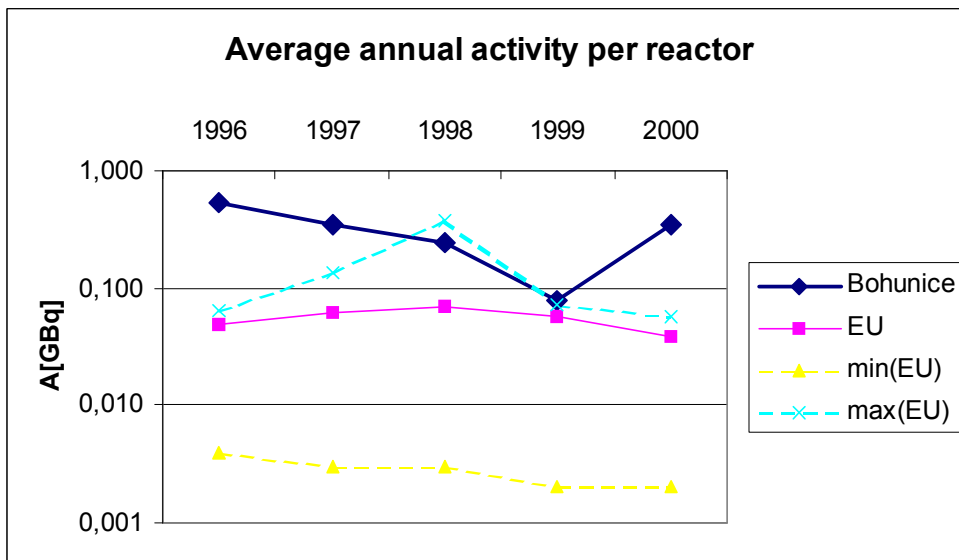
### Comparison of the gas discharges with EU reactors

In order to compare the gas discharges from the Bohunice NPP to EU reactors the data from publication CEPN, 2001: **Liquid and gaseous activity released from pressurized water reactors: International data (1975-2000)** were used. A group of 83 reactors from Belgium, France, Germany and Spain was created and as the period for comparison 5 consecutive years 1996 – 2000 was defined.

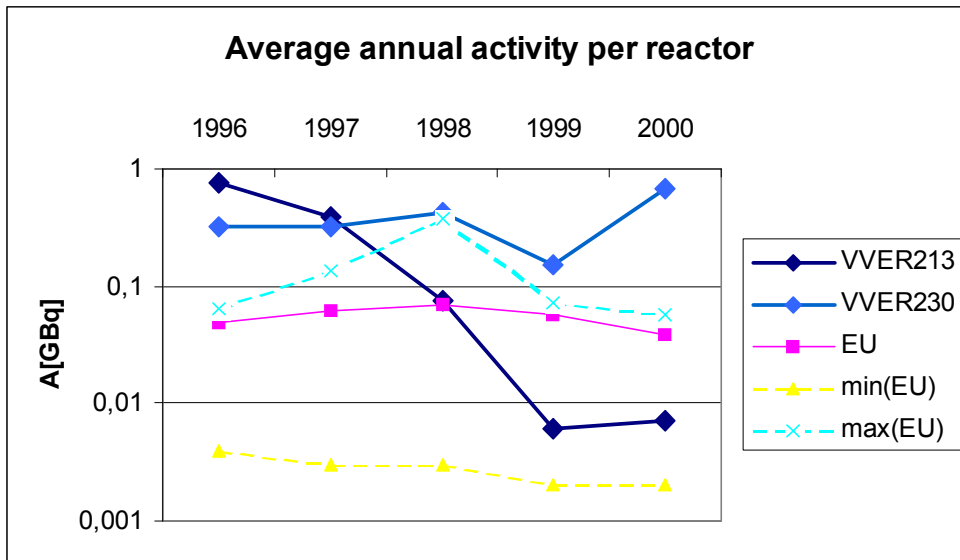
The figure below shows the comparison of the average values of **the noble gas** discharges per reactor unit in given years for Bohunice NPP versus EU reactors. EU(min) and EU(max) are minimum and maximum values of discharges of EU reactors.



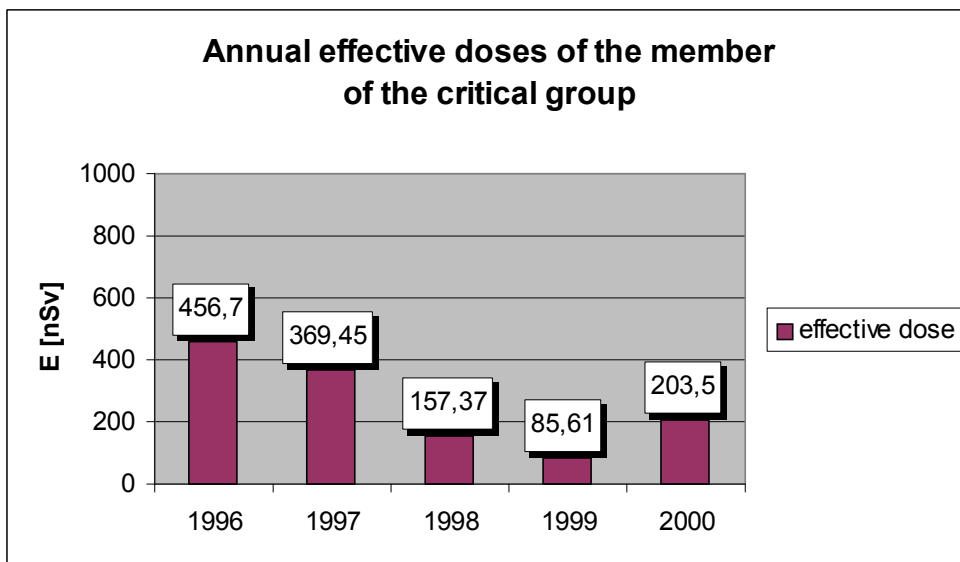
Next graph demonstrates the similar comparison as in previous figure but for **aerosols and halogens**



The third graph contains the average discharged activities of **aerosols and halogens** divided into two items – average discharge from reactor **VVER 230** (Bohunice V1) and **VVER 213** (Bohunice V2)



### Influence of the Bohunice NPP to its surroundings



The last graph contains the annual effective exposures of the member of the critical group in given years calculated from Bohunice NPP gas discharges [3]. The calculated doses are less than 1/1000 of the natural exposures. It is necessary to point out that the contribution from liquid effluents to the total annual exposure of the member of critical group is less than 0,5%. Such exposures lead to the risk values within the interval of statistical errors of present epidemiological studies [4].

## Conclusion

A lot of modifications have been done in order to improve the standard of radiation protection. The modernisation of the gas discharge monitoring system helped to increase the precise of the monitoring concerning to the continual monitoring as well as the laboratory measurements of the discharge samples

Radiation protection at Bohunice NPP has reached the high international standard. The fact was approved by several independent international missions (OSART, WANO, WENRA, ...) and also by the State Health Institute, the Slovak regulatory body in the radiation protection

The operation of Bohunice NPP has negligible influence to its surroundings. The values of gas and liquid effluents move within percents of annual limits given by the regulatory body.

Doses calculated to member of the critical group of the surroundings has been lower than 1/1000 of natural exposure

Results are published in quarter and annual reports and they also are provided to the EC/NEA database

**In the view of gaseous discharges the reactors in Bohunice NPP (or the operation of the reactors) do not differ from reactors in European Union.**

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# RELEASE FROM CONTROL OF INACTIVE MATERIAL FROM DECOMMISSIONING THE ASTRA RESEARCH REACTOR

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## **Abstract**

The Austrian Research Centers Seibersdorf have been operating a 10 MW ASTRA research reactor from 1960 until 1999. After that date, a submission of the intention to decommission the reactor has been provided to the Competent Authorities. After completion of an Environmental Impact Study by the Competent Authorities and modification of the Permissions for Site Use, the reactor finally entered the decommissioning phase in 2003.

Inactive materials from the decommissioning site are expected to be released from control. The procedure for such a release from control agreed upon between the Competent Authorities and ARC Seibersdorf involves a four-step measurement, verification, and certification process detailed in this paper. By September 2003, this four-step procedure has been completed for 16500 kg of steel re-enforced concrete and for 5500 kg of other materials; the release from control of 3000 kg of paraffin and 10000 kg of graphite from the thermal column are planned for the near future.

## **Introduction**

For 39 years, the Austrian Research Centers Seibersdorf had been operating a 10 MW ASTRA research reactor when, after the final shut-down of the reactor in 1999, a submission of the intention to decommission was provided to the Competent Authorities. After completion of an Environmental Impact Study by the Competent Authorities and modification of the Permissions for Site Use, the reactor finally entered the decommissioning phase in

2003. By September 2003, most reactor materials have been evaluated for activation or contamination and their nuclide vectors have been determined.

Large amounts of materials, however, which have not been activated in the course of the reactor's life time and which have not been in contact with unsealed radioactive material can be considered inactive. These inactive materials include the building's furniture and electrical system, building supply and drainage systems other than the reactor operation systems, parts of the experimental setups spatially separated from the reactor core, and the building's structural materials other than the part of the biological shield closest to the core. During the course of the decommissioning work, all these materials will be measured for their radionuclide content and released from control as inactive material, provided that the release protocol mandated by the Competent Authorities has been followed properly.

This paper describes the release protocol for inactive material in the decommissioning of the ASTRA reactor and the measurement methods employed by ARCS and ARC Seibersdorf research to ensure full compliance with Austrian radiation safety regulations. The measurements performed on the various material matrices are described, and the legal procedure, from approval by the site Radiation Safety Officer to independent checks of a subset of the materials by an outside accredited institution to final release from control by the Competent Authorities, is explained in detail.

By September 2003, release from control of several materials has already been authorized by the Competent Authorities, following the four-step procedure outlined in this paper. The measurement methods, materials, and the final release from control are discussed.

### **The ASTRA Research Reactor**

The 10 MW ASTRA research reactor reached first criticality on 29 September 1960. After 39 years of operation, it was finally shut-down on 31 July 1999. The spent fuel elements were removed and shipped-off in 2001. The Environmental Impact Assessment, as mandated by EU and Austrian legislation, was initiated. With its completion, the ASTRA research reactor entered its decommissioning phase in 2003.

### **Release from Control**

Austrian and international regulation mandate that all material in controlled areas be checked and for contamination, activation, or the presence of radioactive sources before it can be permitted into publicly accessible areas. The release from control for reuse, recycling, or final disposal of materials found to be inactive by suitable measurements is an act of the

Competent Authorities and is expected to be finalized by their written authorization by the means of a release certificate.

During a public hearing upon completion of the environmental Impact assessment for the decommissioning of the ASTRA research reactor, a protocol for release from control of materials present in the reactor building, the reactor building structure, and any auxiliary components of the reactor operation systems area was agreed upon between the Competent Authorities, ARC Seibersdorf, and all relevant stakeholders. This procedure involves a four-step protocol, including initial measurements by reactor operations personnel, who are now responsible for decommissioning the reactor. The Radiation Safety Officer or the Radiation Safety Department of ARC Seibersdorf is to assess the methods used by operations personnel, conduct additional measurements, and verify the results. An independent check of the results obtained by operations personnel and the Radiation Safety Officer is provided by an external expert. The results are verified by checking the measurement data and by randomly sampling and measurement 5% of the material to be released from control. Provided the measurement data show compliance with the nationally regulated release levels, the Competent Authorities issue of a release certificate for the materials under investigation, such that they are cleared for reuse, recycling, or final disposal.

The detailed documentation accompanying the release procedure includes the collection and archiving of all measurement data and results, photographic evidence of the materials, the proper signatures of personnel conducting the measurements, the Radiation Safety Officer, and the external expert, and official request for issue of a release certificate together with the release certificate.

All materials released from control have to be voluntarily accepted by the recipient, who issues a properly signed statement to this effect.

### **Materials, Methods, and Measurements**

The release from control of inactive material from the ASTRA research reactor involves various materials, such as concrete, steel and graphite, paraffin, cables and pipes, electronics, furniture, and flooring.

To investigate the potential of the various materials for containing radioactive sources, for being activated, or for being contaminated, an investigation of their modes of use during their lifetime in the reactor building is conducted. For activation products, the integral neutron fluence at the material's location is estimated, and the material composition is determined. Contaminations are only expected to be present in reactor components and in tools and

machinery previously used for handling active material, such as fuel or experimental equipment. Any other areas in the reactor building and materials have been checked continuously for surface contamination, even during regular reactor operation. Given these considerations, a list of expected nuclide vectors for the various materials is generated.

Various different measurement methods and techniques are employed, depending on the expected radionuclide vectors. Representative samples are taken of all the materials for alpha and beta emitting nuclides. Chemical preparation is performed, after which alpha spectrometry is employed for alpha emitters. Beta emitters are examined by liquid scintillation counting, or on a gas counter. Gamma emitting radionuclides in large amounts of material are detected, identified, and quantified by in-situ gamma spectrometry. Surface contamination is determined by hand-held gas counters or plastic scintillators, or by swipe test sampling.

## **Results**

By September 2003, various materials have been released from control by the procedure outlined above. Steel re-enforced concrete used as mobile shielding material of an amount of 16500 kg and 5500 kg of furniture, cables, small machinery, and electronics have been released for recycling. Most of that material, however, has been turned over for final waste disposal.

The materials currently under investigation include 3000 kg of paraffin which was used as mobile neutron shielding and 10000 kg of graphite from the reactor's thermal column which was used to generate thermal neutrons for activation experiments. Of the total amount of paraffin, about 50 % have been measured. No contamination has been found so far. About 30 % of the thermal column's graphite has already been released. However, completion of its dismantling is not expected before December 2003, such that final release of all the graphite is not possible before early in 2004.

## **Conclusions**

After 39 years of operation, the ASTRA reactor at the Austrian Research Centers Seibersdorf is currently being decommissioned. Various inactive materials from the reactor building and structure are being released from control by employing a four-step procedure, agreed upon by the Competent Authorities, ARC Seibersdorf, and all relevant stakeholders during a public hearing in December 2002.



The materials released to date include 16500 kg of steel re-enforced concrete from mobile shielding material and 5500 kg of furniture, cables, small machinery, and electronics. About 10000 kg of graphite from the thermal column and 3000 kg of paraffin from mobile neutron shielding.

After completion of the dismantling and release from control of the thermal column graphite, the next and biggest step in the decommissioning process will be approached. The dismantling, separation of active and inactive regions, and release from control of inactive material from the biological shield will be started.

# CLEARANCE (FREE RELEASE) OF SLIGHTLY RADIOACTIVE MATERIAL IN ARC SEIBERSDORF

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## **Abstract**

A new radiation protection law designed to implement the recommendations of the 29/96/EURATOM directive has recently been adopted in Austria. Austrian Research Centers GmbH – ARC, Seibersdorf, is in the process of adapting its clearance operations accordingly.

The conditions to be fulfilled prior to clearance derive from the 10  $\mu\text{Sv/a}$  additional dose concept. Activity concentration limits for each radionuclide result. In order to assure compliance with the rather restrictive values sophisticated free release measurements are called for. International experience has shown that acceptable results in terms of quality, throughput, and price, are practically only achievable by using a purpose built automated measurement unit, working on the principle of total  $\gamma$  counting. Calibration and measurement protocol play a key role in obtaining reliable results. The issue of homogeneity plays a central role in the clearance of objects exceeding regulatory mass (surface) limits.

Experience with installation and operation of a free release measurement unit in ARC Seibersdorf are discussed.

## **Introduction**

Clearance (free release) of slightly radioactive material is defined<sup>1</sup> as ‘removal of radioactive materials or radioactive objects within authorized practices from any further regulatory control by the regulatory body’.

Clearance is based upon the realization that the radioactivity of some materials is so low as to render them harmless. In other words, the dose associated with exposure to these materials is trivial. The currently recommended value for the trivial dose is 10 - 100  $\mu\text{Sv/a}$ <sup>2</sup>,

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<sup>1</sup> Radioactive Waste Management Glossary, IAEA, 2003.

<sup>2</sup> Safety Series 89, IAEA, 1988

or 10  $\mu\text{Sv/a}$  for a single clearance practice<sup>3</sup>. These recommendations have been adopted for use in the EU<sup>4</sup>.

The practical application of this concept necessitates the calculation of activity concentration (volume or surface) limits for each nuclide from the trivial dose and *vice versa*. The results critically depend on the assumed exposure scenario, as well as on a number of other variables, such as the physical state of the material to be free released, the disposal pathway, *etc.* However, the accuracy and validity of such calculations are in question, *e.g.*, there are indications that activity concentration limits thus calculated might be an order of magnitude too conservative<sup>5</sup>.

### **Situation in Austria**

Similarly to Germany and Switzerland, a new radiation protection law designed to implement the recommendations of the 29/96/EURATOM directive has recently been adopted in Austria<sup>6</sup>. A new radiation protection ordinance will follow shortly. Hence, a novel legislative framework for clearance has been created. Of special importance for clearance is the transition from the previously used 10  $\mu\text{Ci/m}^3$  (0,4 Bq/g at a density of 1  $\text{g/cm}^3$ ) activity concentration limit to the set of nuclide specific values.

Austrian Research Centers GmbH – ARC, Seibersdorf, as the only facility for collection, treatment, conditioning, and interim storage of low and intermediate level waste in Austria, is in the process of adapting its clearance operations accordingly.

### **Instrumentation**

In order to assure compliance with the rather restrictive activity concentration limits sophisticated free release measurements are called for. These can be performed either by hand or automatically and either spectroscopically or by total  $\gamma$  counting. International experience has shown that the necessary quality and throughput at a justifiable price is practically only achievable by using an automated total  $\gamma$  counting measurement unit.

These units typically employ shielded large area plastic scintillator detectors in a  $4\pi$  geometry. Because of operation in the vicinity of the detection limits, calibration and measurement protocol play a key role in obtaining reliable results. In particular, the effects of background, natural radioactivity of the measured material, and the shielding caused by the

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<sup>3</sup> TECDOC-855, IAEA, 1996

<sup>4</sup> Directive 96/29/EURATOM, EC, 1996

<sup>5</sup> S. Menon, private communication, 2003

measured material have to be considered. Calibration is typically performed with a key nuclide, *e.g.*,  $^{60}\text{Co}$ ,  $^{137}\text{Cs}$ , yielding a set of calibration factors (one for every detector). The measured activity is considered ‘key nuclide equivalent’ and activities of all nuclides of interest are evaluated using a measured or an estimated nuclide vector. Separate calibrations are performed for different types of material, *e.g.*, pipes, sheet metal, insulation, concrete rubble, *etc.*

### **Homogeneity**

The immediate result of a clearance measurement using an automated total  $\gamma$  counting measurement unit are the count rates of the  $n$  detectors,  $\{Z_i\}_{i=1,2,\dots,n}$ . The calculation of the average nuclide specific activity concentration of the material using the detector calibration factors, the nuclide vector, and the mass (surface) of the material is straightforward. However, the regulatory mass (surface) limit of 300 kg (1000 cm<sup>2</sup>) presents a problem for measuring material with larger mass (surface). This can be dealt with in a conservative way by using the regulatory mass (surface) limit irrespective of the actual mass (surface) of the material, possibly leading to not clearing ‘clearable’ material. Hence, a more sophisticated approach relying on the concept of homogeneity of activity concentration distribution deserves closer scrutiny. The calculation of the activity concentration distribution in the material from  $\{Z_i\}_{i=1,2,\dots,n}$  is not trivial. The practical application of the homogeneity concept is further complicated by the lack of relevant regulatory limits.

### **Implementation in ARC**

A modern total  $\gamma$  counting clearance measurement unit has recently been installed in a completely refurbished hall at ARC Seibersdorf. The background in the hall is on the order of 100 nSv/h. The unit features 10 plastic scintillator detectors measuring 50 x 50 x 5 cm each and an integral scale with a 1000 kg weight limit. It is intended primarily for handling material in standard 200-L drums but can be used for differently packaged material as well, including the option to measure long items with the door open.

The kinds of material to be measured and cleared include excavated soil, concrete rubble, empty 200-L drums, decay waste, *etc.* A separate calibration will be performed for each material type by using identical or similar inactive material with a  $^{60}\text{Co}$  or a  $^{137}\text{Cs}$  source imbedded in various positions.

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<sup>6</sup> There are other ways of dealing with clearance, *e.g.*, the case-by-case approach in the USA or the 0,4 Bq/g activity concentration limit in use in the UK.

# OPERATIONAL CONTROL OF MATERIAL RELEASE AND DISCHARGES FROM NUCLEAR POWER PLANT

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## Introduction

The operational control of radioactive materials during atmospheric release and aquatic discharge from nuclear power plant is a licensing criterion for NPPs. Originally at the Paks NPP the release control was based on activity limits for four groups of elements. These groups were noble gases, long living radioaerosols, radioiodine and radiostrontium for atmospheric release and specified activity limit for beta emitters, strontium and tritium for aquatic discharge into Danube. These groups were controlled with proper sampling and/or measuring instrumentation. The limit for atmospheric release was given as a 30-day moving average, for liquid discharges the annual limit was stipulated.

The new release and discharge limitation system is based on the environmental dose limitation. The dose constraint for Paks NPP is 90  $\mu\text{Sv}/\text{year}$  of the critical group for all release pathways and the investigation dose limit is equal to 27  $\mu\text{Sv}/\text{year}$ . The regulation did not subdivide the dose limit for atmospheric and liquid components but for operational control subdivision of dose limits for atmospheric release and aquatic discharge and shorter time period (one day-one month) seems to be useful. The subdivision can be based on past release data and/or previous activity limits. To satisfy dose below the investigation dose limit there should be a proper operation control level for each separately measured component and pathway belonging to reasonable time interval significantly shorter than one year. The main task of the NPP staff is elaboration of reasonable control levels and reference time intervals for different radionuclide and element groups to be used in operational control.

## Environmental dose values due to former atmospheric release and aquatic discharge limits

The former stack release limit (until 2002) was given for four radioactive material groups normalized to 1000 MW electrical capacity. We have calculated the dose for the critical group of population due to continuous atmospheric release equal to the limit (Table 1). The atmospheric dose conversion factor for each radionuclide was calculated using locally measured meteorological data (120 m high meteorological tower) during 5 years. For further calculation the highest dose conversion factor of the 5 year set was used. For dose conversion factors of aquatic discharge hydrological data of river Danube were applied. The dose conversion factors for a group of nuclides were taken equal to the highest one in the group. The total dose in this case is 200  $\mu\text{Sv}/\text{year}$  that is much higher than the full (atmospheric and aquatic pathways) investigation dose limit, i.e. 27  $\mu\text{Sv}/\text{year}$ .

Due to regulation for aquatic pathway discharge limit for beta emitters (except tritium) 14.8 GBq/year, for Tritium 30 TBq/year (Table 2). Taking into account the actual mean composition of the discharge the main dose components (except Tritium) consist of Co-58,

Co-60, Cs-134 and Cs-137. The upper value of the annual dose due to mixture of beta emitters (except Tritium) in aquatic discharge can be calculated using conservative assumption. For this case we have to take the highest dose conversion factor, i.e.  $1.3E-7$  Sv/GBq for Cs-134. Using this factor the dose values for former discharge limits are given in Table 2.

Table 1. Former atmospheric release limits and calculated doses

Group	Release limit for 1000 MW [Bq/day]	Release limit for 1760 MW [Bq/y]	Dose conversion factor [nSv/Bq]	Dose [nSv/y]	Dose fraction [%]
noble gases	1.90E+13	1.22E+16	3.1E-12 (Kr-88)	3.78E+04	20
I-131	1.10E+09	7.07E+11	1.2E-07	8.48E+04	42
Aerosols	1.10E+09	7.07E+11	1.1E-07 (Cs-134)	7.77E+04	38
Sr-90	5.60E+04	3.60E+07	2.4E-07	8.63E+00	< 0.01
Total				2.00E+05	

Table 2. Dose components due radionuclides in case of aquatic discharge limit

Group	Discharge limit [Bq/year]	Dose conversion factor [nSv/Bq]	Dose [nSv/year]
Beta emitters	1.48E+10	1.3E-07*	1924
Tritium	3.00E+13	3.1E-12	93
Total			2017

\*dose conversion factor for Cs-134

### Environmental dose values due to measured atmospheric release and aquatic discharge

The measured stack release for two stacks in 2002 and calculated doses for critical group of population are given in Table 3. The dose fractions of different components are shown in Fig 1. The full dose due to atmospheric release is below  $0.1 \mu\text{Sv/year}$ .

The measured aquatic discharge in 2002 and calculated doses for critical group of population are given in Table 4. The dose fractions of different components are shown in Fig. 1. Significant part (30%) of the dose caused by Tritium, the other components can be estimated using gross beta counting. In this case the dose using the highest dose conversion factor ( $1.3E-07$  nSv/Bq for Cs-134) the computed dose is 325 nSv/year, i.e. about two times higher than using nuclide specific measurement. The full dose due to aquatic release is below  $0.3 \mu\text{Sv/year}$ .

### Subdivision of investigation dose limit for operational control

The dose equivalent of the earlier declared atmospheric release limit in activity units is equal to 200  $\mu\text{Sv}/\text{year}$  (Table 1) which is much higher than the total (atmospheric + aquatic) investigation dose limit, i.e. 27  $\mu\text{Sv}/\text{year}$ . In the same time the dose equivalent of the earlier declared aquatic discharge limit in activity units is equal to 2  $\mu\text{Sv}/\text{year}$  which is much lower than the full investigation dose limit. Taking into account that the possible dynamics of aquatic discharge is much less than that for aquatic discharge introduction of a virtual subdivision of investigation dose limit for aquatic discharge equal to 2  $\mu\text{Sv}/\text{year}$  can be justified. There is an additional – very low intensity – release point: stack of the laboratory building for sample preparation. For this point investigation dose limit 1  $\mu\text{Sv}/\text{year}$  has several orders of magnitude of safety factor. Finally the "slice" for two ventilation stack together remains 24  $\mu\text{Sv}/\text{year}$  investigation dose limit, 12  $\mu\text{Sv}/\text{year}/\text{stack}$ . This subdivision is necessary only for comparison of control level to be used with "slice" of investigation dose limit.

Table 3. Measured stack release for two stacks in 2002 and calculated doses

Group	Nuclide	Release [Bq/year]	Dose conversion factor [nSv/Bq]	Dose [nSv]	Dose fraction %	Subtotal	
						dose [nSv]	Fraction %
noble gases	Ar-41	1.13E+13	2.0E-12	22.5	26.48	27.1	31.9
	Kr-88	1.33E+12	3.1E-12	4.1	4.85		
	Xe-133	1.62E+12	4.4E-14	0.1	0.08		
	Xe-135	1.10E+12	3.7E-13	0.4	0.48		
Tritium	HTO form	6.06E+12	5.2E-13	3.2	3.70	3.2	3.7
C-14	CO <sub>2</sub> form	5.00E+10	7.1E-10	35.5	41.73	35.5	41.7
Aerosols	Mn-54	1.32E+08	9.0E-09	1.2	1.40	9.1	10.7
	Co-58	1.25E+08	4.2E-09	0.5	0.61		
	Fe-59	2.37E+07	8.1E-09	0.2	0.23		
	Co-60	1.27E+08	3.8E-08	4.8	5.67		
	Ag-110m	4.72E+07	1.9E-08	0.9	1.05		
	Cs-134	2.23E+06	1.1E-07	0.2	0.29		
	Cs-137	1.44E+07	8.7E-08	1.2	1.47		
Iodine	I-131	8.48E+07	1.2E-07	10.2	11.96	10.2	12.0
Total						85.1	100.0

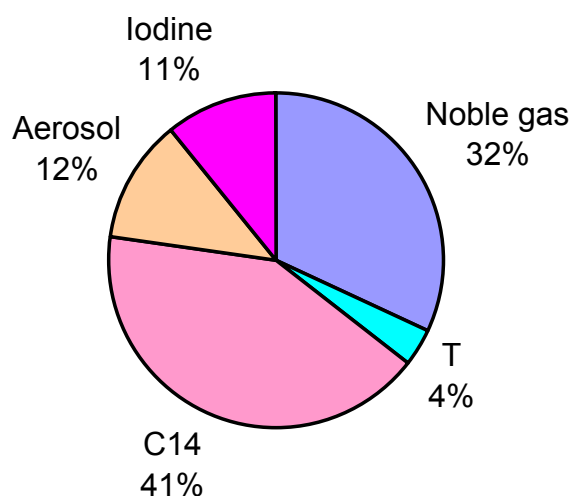


Fig. 1. Dose fraction of different components due to measured atmospheric release in 2002

Table 4. Aquatic discharge in 2002 and calculated doses

Nuclide	Release [Bq/year]	Dose conversion factor [nSv/Bq]	Dose [nSv/year]	Dose fraction [%]
C-14	1.0E+08	3E-08	3.1	2.0
Sr-90	1.9E+06	4.2E-08	0.1	0.1
Fe-55	1.7E+08	2.1E-09	0.4	0.2
Mn - 54	4.2E+08	9.1E-09	3.8	2.4
Co - 58	2.8E+08	2.8E-08	7.9	5.1
Fe-59	4.0E+07	4.0E-08	1.6	1.0
Co - 60	1.1E+09	9.3E-08	103.7	66.6
Nb - 95	4.4E+06	4.3E-08	0.2	0.1
Ag - 110	6.2E+07	4.6E-09	0.3	0.2
I - 131	8.0E+06	3.4E-08	0.3	0.2
Cs - 134	1.0E+08	1.3E-07	13.0	8.3
Cs - 137	2.2E+08	9.8E-08	21.5	13.8
subtotal	2.5E+09		155.8	100.0
Tritium	2.2E+13	3.1E-12	67.8	–
Total			223.6	–

For calculation of upper value of the operational control of the atmospheric release the investigation dose limit can be divided into smaller parts. We can divide release into nuclide groups (noble gases, Tritium, C-14 and aerosols) due to types of the measuring equipment and in next step divide this value by two to obtain the limit for each ventilation stack separately. Results of these calculations are given in Table 5.



Table 5. Calculated investigation dose and release limits for integrating time periods and one stack

Group	Dose fraction	Dose [nSv/year]	Dose conversion factor [nSv/Bq]	Calculated limit			
				[Bq/year]	time unit (t.u.)	[Bq/t.u.]	Bq/t.u. for 1 stack
noble gases	31.9	7655	3.1E-12	2.47E+15	day	6.77E+12	3.38E+12
Aerosols	10.7	2565	1.1E-07	2.33E+10	day	6.38E+07	3.19E+07
Iodine	12.0	2877	1.2E-07	2,40E+10	day	6.57E+07	3.28E+07
Tritium	3.7	889	5.2E-13	1.71E+15	month	1.42E+14	7.12E+13
C-14	41.7	10014	7.1E-10	1.41E+13	month	1.18E+12	5.88E+11
Total	100.0	24000					

Due to mentioned calculations the preliminary release limit for one stack (Table 5).

- noble gases – 3.4E+12 Bq/day (3400 GBq/day)
- Aerosols – 3.2E+07 Bq/day (32 MBq/day)
- Iodine – 3.3E+07 Bq/day (33 MBq/day)
- Tritium – 7.1E+13 Bq/month (71000 GBq/month)
- C-14 – 5.8E+11 Bq/month (580 GBq/month)

### Operational control level for atmospheric release

Operational control level is introduced to indicate all irregular events leading to excess release of radioactive material. These levels should be high enough to indicate only the unusual events and low enough in comparison with the proper "slice" of the investigation dose limit to ensure release below investigation level.

Five continuous sampling and measuring channels are installed at both ventilation stacks (Table 6). (Units 1 and 2 are connected to one stack, units 3 and 4 to the second one.)

Table 6. Sampling and measuring channels at ventilation stacks

Channel	Measuring period	Averaging period [day]
Noble gas	10 minutes	1 (moving average)
Aerosols	10 minutes	1 (moving average)
Iodine (I <sub>2</sub> +organic)	10 minutes	1 (moving average)
Tritium	2 weeks	1 month
C-14	2 weeks	1 month

To find the optimal control level we studied the dynamics of release rates in 2002.

## Analysis of release dynamics

Calculated in previous chapter investigation release limits are valid only in case of constant release rates. These values should be compared with dynamics of real release rate. On the base of the measured release rate per day or month the recommended operational control levels can be determined. For operational control levels can be fixed at levels occurring only several times a year in each measuring channel to indicate unusual release rate for NPP staff. These levels should be significantly less than investigation release limits.

### *Noble gases*

Above 300 GBq/day 5 cases were found (321, 326, 371, 338 and 358 GBq/day). Fig. 2 shows release rate of stack Units 1 and 2 for period July-December of 2002. The other block and time periods show similar character. It is evident that the release rate/day has several sharp peaks connected with technological events. Using 300 GBq/day operational release control levels such events can be detected for internal investigation of the event. This operational control value is more than 10 times lower than the time proportional part of the above-calculated investigation release value, i.e. 3400 GBq/day.

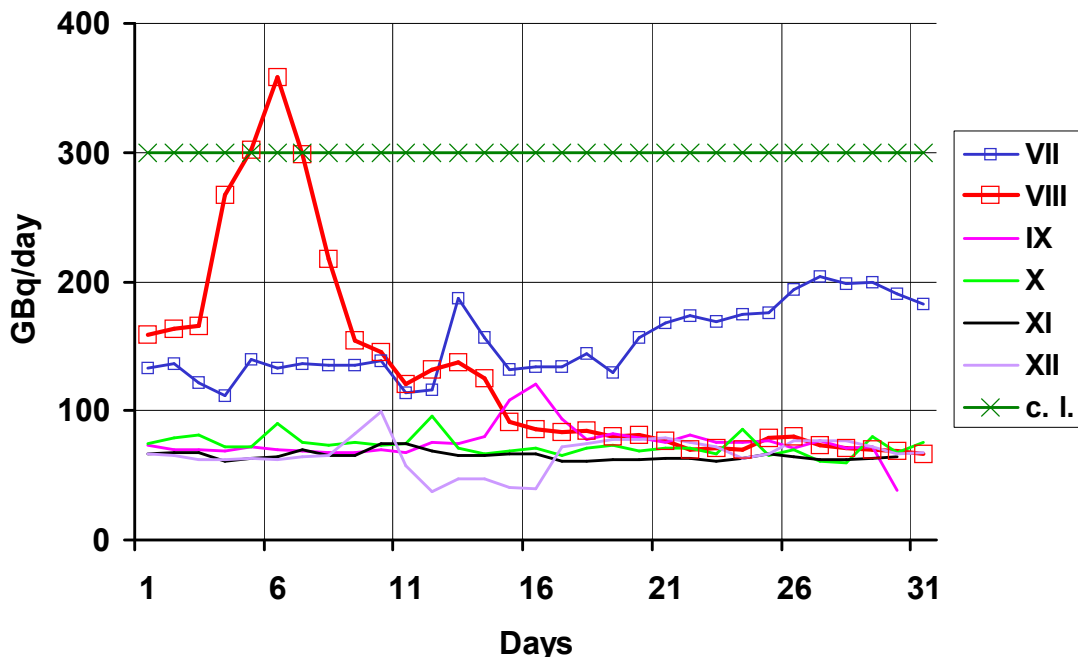


Fig. 2. Noble gas release rates of stack of Units 1 and 2 for period July-December 2002 and its control level (c. l.)

### *Aerosols and elementary Iodine*

Release of Iodine may occur in different form, i.e. in aerosol, elementary (vapor) and organic form. The measuring equipment measures the aerosol and the elementary+organic forms separately, therefore separation of the operational control level for aerosol and (elementary) iodine components seems to be reasonable.

In aerosol component above 1 MBq/day 5 cases were found (1.2, 1.3, 1.3, 1.4 and 2.6 MBq/day). Fig. 3 shows release rate of stack Units 3 and 4 for period July-December of 2002. The other block and time periods show similar character. It is evident that the release rate/day has several sharp peaks connected with technological events. Using 1.2 MBq/day

operational control levels such events can be detected for internal investigation of the event. This operational control level is more than 25 times lower than the time proportional part of the above-calculated investigation release value/stack, i.e. 32 MBq/day.

In elementary Iodine component above 5 MBq/day 3 cases were found (7.1, 7.1 and 5.4 MBq/day). Fig. 4 shows release rate of stack Units 1 and 2 for period July-December of 2002. The other block and time periods show similar character. It is evident that the release rate/day has several definite peaks connected with technological events. Using 5 MBq/day operational release control level such events can be detected for internal investigation of the event.

### *Tritium*

Monthly release rate of Tritium for both stacks separately is given in Fig. 5. The Tritium release rate per months is in the range 100-600 GBq/month. In this case the operational control level 800 GBq/month seems to be reasonable for operational control level as release of this radionuclide is not connected with any technological event. Due to smooth time dependence of release the ratio of investigation level (71000 GBq/month) to operational one is very high, about 90.

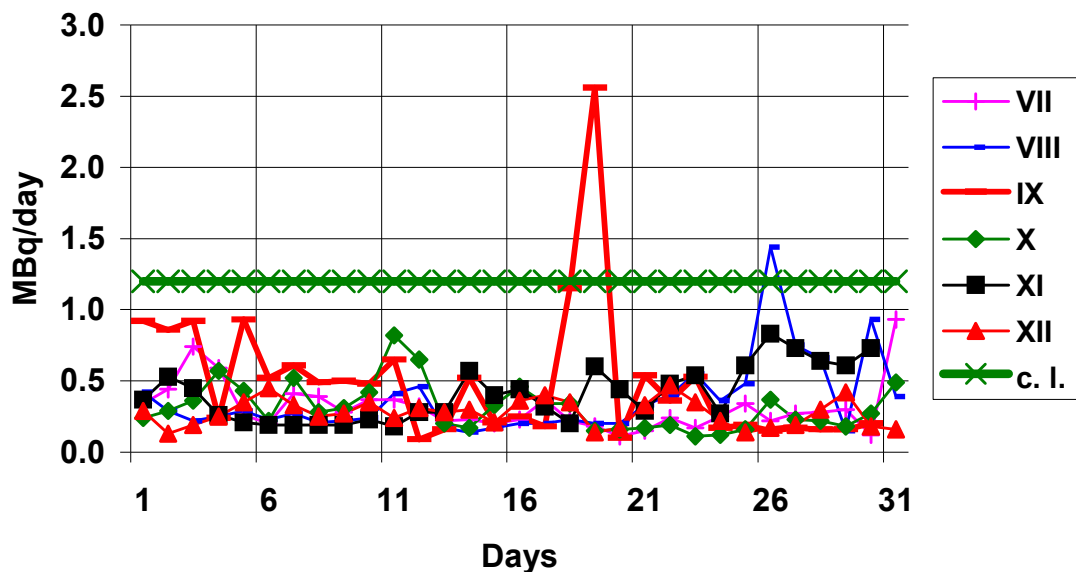


Fig. 3. Aerosol release rates of Units 3 and 4 for period July-December 2002 and its control level (c. l.)

### *C-14*

Monthly release rate of C-14 for both stacks separately is given in Fig. 6. The C-14 release rate per months is in the range of 20-50 GBq/month. In this case the operational control level 60 GBq/month seems to be reasonable for control level as release of this radionuclide is not connected with any technological event.

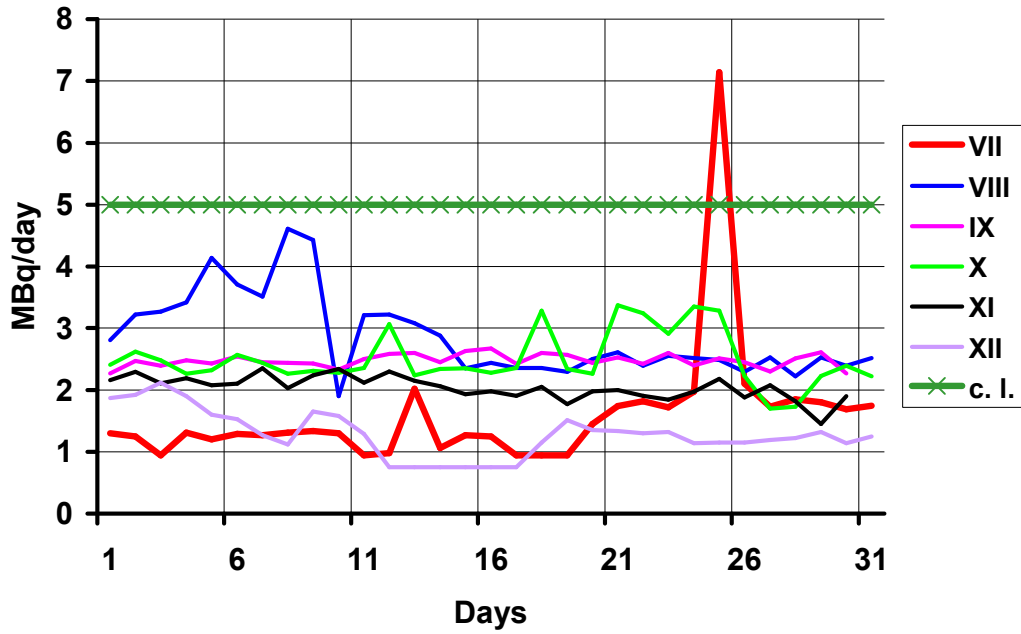


Fig. 4. Iodine release rates of Units 1 and 2 for period July-December 2002 and its control level (c. l.)

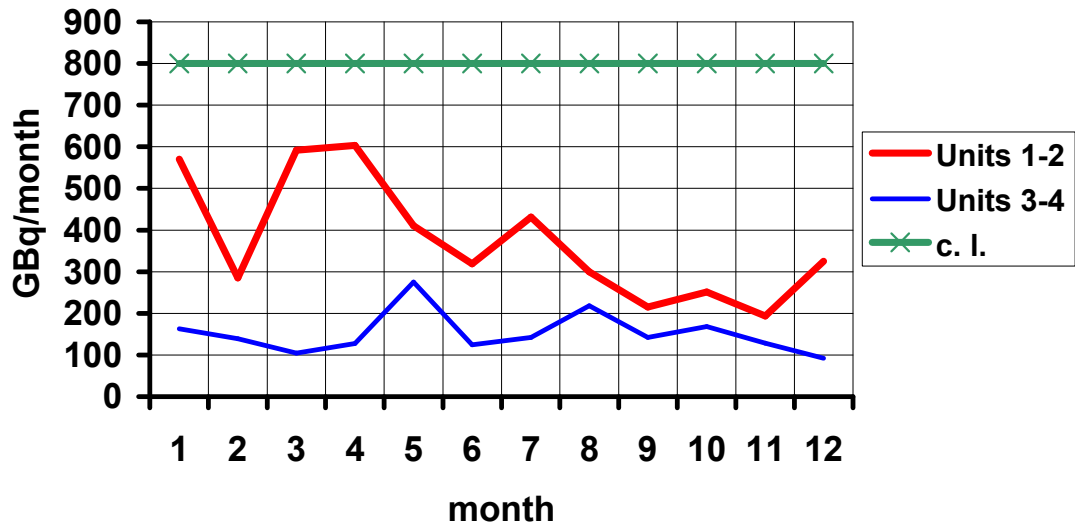


Fig. 5. Tritium release rate in 2002 and its control level (c. l.)

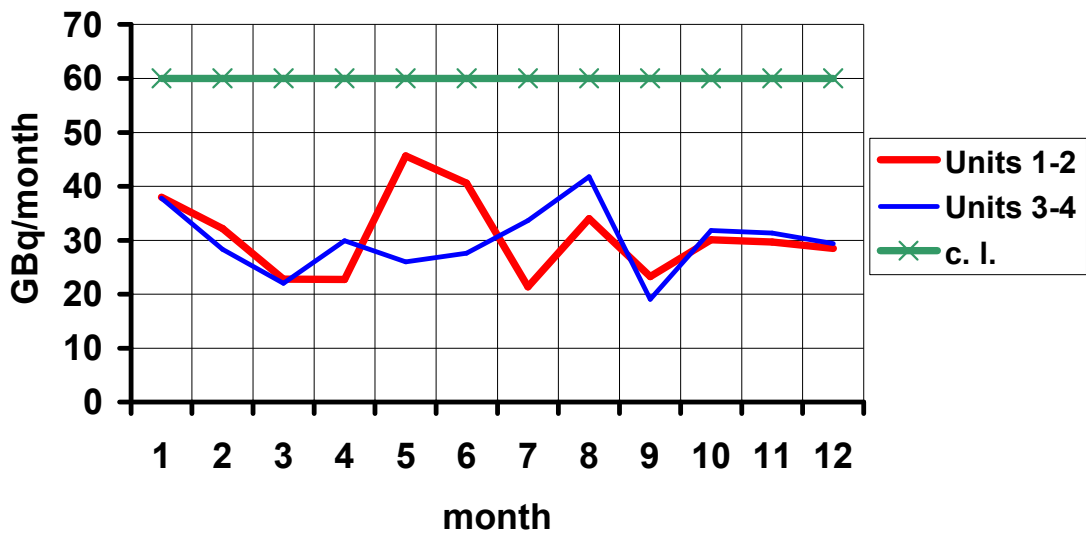


Fig. 6. C-14 release rate in 2002 and its control level (c. l.)

*Liquid discharges*

Activity of liquid discharges is controlled before discharge and therefore definition of control level has much less importance than in case of atmospheric releases. Dynamics of discharged activity is less wide than in most cases of atmospheric releases (Fig. 7.). For controlling the discharge the dose resulted of discharge can be used. On Fig. 8 dose due to components and the total dose is shown. The doses are below 50 nSv/month, so application of dose control level 100 nSv/month seems to be reasonable.

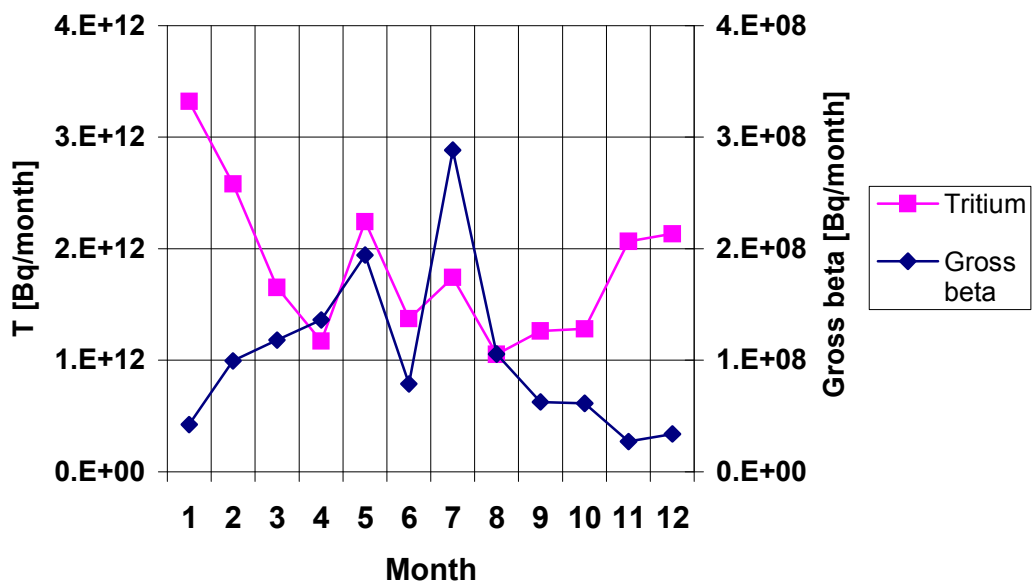


Fig. 7. Discharge of Tritium and beta emitting radionuclides in 2002

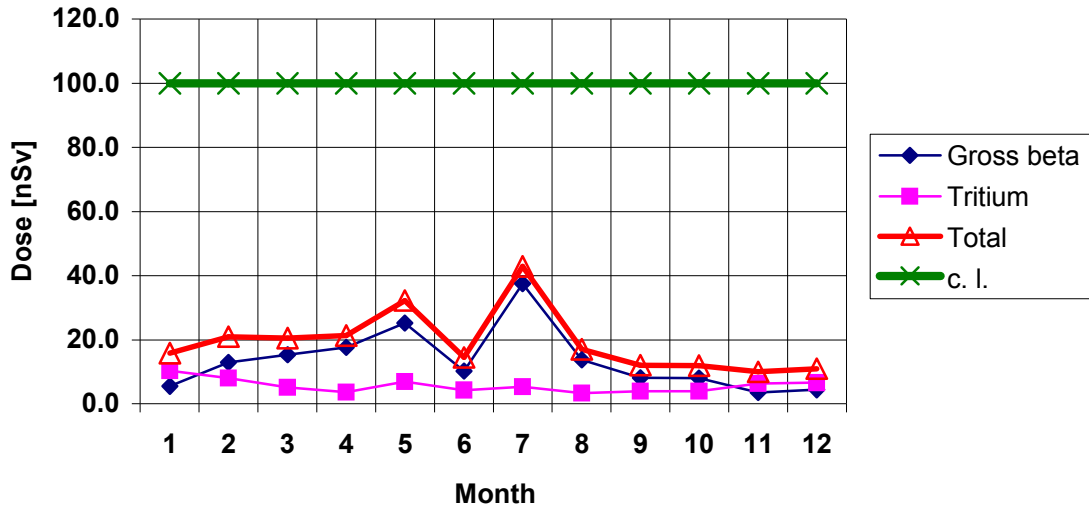


Fig. 8. Dose due to discharge of Tritium and beta emitting radionuclides and total dose due to discharge in 2002

### Conclusions

Operational control levels are based on measured daily or monthly release rates. In case of noble gases, aerosols and iodine the daily release rates have several sharp peaks per year. Operational control levels give opportunity to detect these peaks for internal investigation purposes. Investigation release limits were received from overall investigation release limit based on measured dose fraction of different component. Dose due to operational release control levels in case of constant release rate is 15 times lower (Table 7) than the investigation dose limits giving very high safety factor.

Table 7. Summary of operational control levels and investigation release limits

Group	Nuclide	dose conv. factor Sv/GBq	control level			investigation limit		
			GBq/stack/t.u.	time unit (t.u.)	dose $\mu$ Sv/year 2 stacks	release [Bq] 2 stacks	time unit (t.u.)	dose $\mu$ Sv/year 2 stacks
noble gases	Kr-88	3.10E-12	300	day	0.7	6.77E+12	day	7.7
iodine	I-131	1.20E-07	0.005	day	0.5	6.57E+07	day	2.9
aerosols	Sr-90	2.40E-07	0.0012	day	0.2	6.38E+07	day	2.5
C-14	<sup>14</sup> CO <sub>2</sub>	9.20E-11	60	month	0.15	1.18E+12	month	10.0
T	HTO	5.20E-13	800	month	0.01	1.42E+14	month	0.9
Labor. building	all	n. a.	0.02 $\mu$ Sv	month	0.24 (1 stack)	n. a.	n. a.	1.0
liquid discharge	gr. beta, T	n. a.	0.1 $\mu$ Sv	month	1.2 (dis-charge)	n. a.	n. a.	2.0 (dis-charge)
Total					3.0			27.0

## Safety Study of PET Complex of Cyclotron Center of Slovak Republic

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### *Introduction*

The Cyclotron center of the Slovak Republic (CC SR) will be built in two buildings: pavilion I (PET complex) and pavilion J (cyclotron DC72). CC SR is located in area of Slovak Institute of Metrology, Bratislava (see Fig.1).



Fig. 1. Locality of Cyclotron Center of Slovak Republic in Bratislava

The PET Complex of CC SR consists of three main technological parts: cyclotron for production of positron radionuclides, radiochemical laboratories and nuclear medicine department.

### **Cyclotron for production of positron radionuclides**

The cyclotron for the PET Complex is a 18/9 MeV proton/deuteron accelerator (CYCLONE 18/9 made by IBA, Belgium) for production of radionuclides. Main specifications of the CYCLONE 18/9 are given in Tab.1.

Tab.1 Cyclotron CYCLONE 18/9 specifications

Ions	- extracted	H <sup>+</sup> , D <sup>+</sup>
	- accelerated	H <sup>-</sup> , D <sup>-</sup>
Energy (fixed)	- protons	18 MeV
	- deuterons	9 MeV
Performed extracted intensity on target	- protons	80 μA
	- deuterons	40 μA
Number of target ports		8
Simultaneous extracted beam		2
External beam, intensity on target	- protons	15 μA
	- deuterons	7 μA

Targets planned in the PET Complex are in Tab.2.

Tab.2 Targets

Radio-nuclide	Activity per production run [GBq]	Activities per year [GBq]	Target form
<sup>18</sup> F	111	55500	H <sub>2</sub> O
<sup>11</sup> C	37	9250	N <sub>2</sub> gas
<sup>13</sup> N	3.7	925	H <sub>2</sub> O
<sup>15</sup> O	37	9250	N <sub>2</sub> gas

A cyclotron vault with maze is located under ground near pavilion I on side in direction to pavilion J.

### Radiochemical laboratories

Radiochemical laboratories consist of GMP zone for radiopharmaca production, quality control and laboratories for research of radiopharmaca. Planned production of radiopharmaca is given in Tab.3.

Tab.3 Handled activities in radiochemical laboratories

Radiopharmaca	Handled activity per day [GBq]
<sup>11</sup> C-raclopride	1,85
<sup>13</sup> N-amonia	1,48
<sup>15</sup> O-CO <sub>2</sub> (also H <sub>2</sub> O)	50,0
<sup>18</sup> FDG	2x60

### Nuclear medicine

Department of nuclear medicine is assigned for examination of patients with gamma cameras for positron emission tomography (PET) and with coincidence camera PET/SPECT for PET and single photon emission tomography (SPECT). Planned numbers of patients and amount of radiopharmaca for PET and SPECT examinations are shown in Tab.4 and Tab.5.



Tab.4 Handled activities for the positron emission tomography (PET)

Radiopharmaca	T <sub>1/2</sub>	Activity per week [GBq]	Patients per week
<sup>11</sup> C-Raclopride	20.3 m	1,85	5
<sup>13</sup> N-amonia	10 m	1,48	4
<sup>15</sup> O-CO <sub>2</sub> (+ H <sub>2</sub> O)	123 s	50,0	5
<sup>18</sup> FDG	110 m	600	60

Tab.5 Handled activities for the single photon emission computerized tomography (SPECT)

Radio-nuclide	T <sub>1/2</sub>	Photon energy [MeV]	Activity per week [GBq]	Patients per week
<sup>123</sup> I	13,3 h	0,159, 0,53	2	5
<sup>111</sup> In	67,3 h	0,19, 0,95	0,3	2
<sup>67</sup> Ga	78,3 h	0,09–0,89	0,6	2
<sup>81m</sup> Kr	13 s	0,191	90	15
<sup>201</sup> Tl	72,9 h	0,03–0,167	0.2	2
<sup>99m</sup> Tc	6 h	0,14	6,6	30

### ***PET Cyclotron Center Safety Study***

The scope of the Safety Study for the PET Complex is based on the radiation protection program, published in Safety Series No.102 :

- A. Organization and management committed to safety and ALARA
- B. Successful personnel selection and training
- C. Effective occupational radiation control
- D. Effective public radiation control
- E. Effective emergency planning and preparedness
- F. Implemented quality assurance

The items A, C and D have been elaborated. The rest of the Safety Study will be finished in Jun 2004 (the date of planed activation of radiation sources in the PET complex).

### **Provision of physical barriers in PET complex**

Radioactive sources in PET complex :

- Prompt and isotopic sources in cyclotron vault
- Positron radionuclides in radiochemical laboratories
- Patients with administered radiopharmaca for PET and SPECT examinations
- Radioactive aerosols and gases
- Liquid radioactive waste storage
- Treasures with radioactive sources

### **Cyclotron vault**

The cyclotron CYCLONE 18/9 with target for the <sup>18</sup>F production was simulated by the Monte Carlo code MCNP4B. Angular distribution of neutron flux density around the target for the <sup>18</sup>F production was calculated by the MCNPX code, version 2.3.

The calculated angular distribution of neutron flux density was used for simulation of neutron energy spectra at Cyclone 18/9. Comparison of neutron dose equivalent around the cyclotron with experiments (measured by IBA) is shown on Fig.1.

The distance of 0.5 m from the surface of CYCLONE 18/9

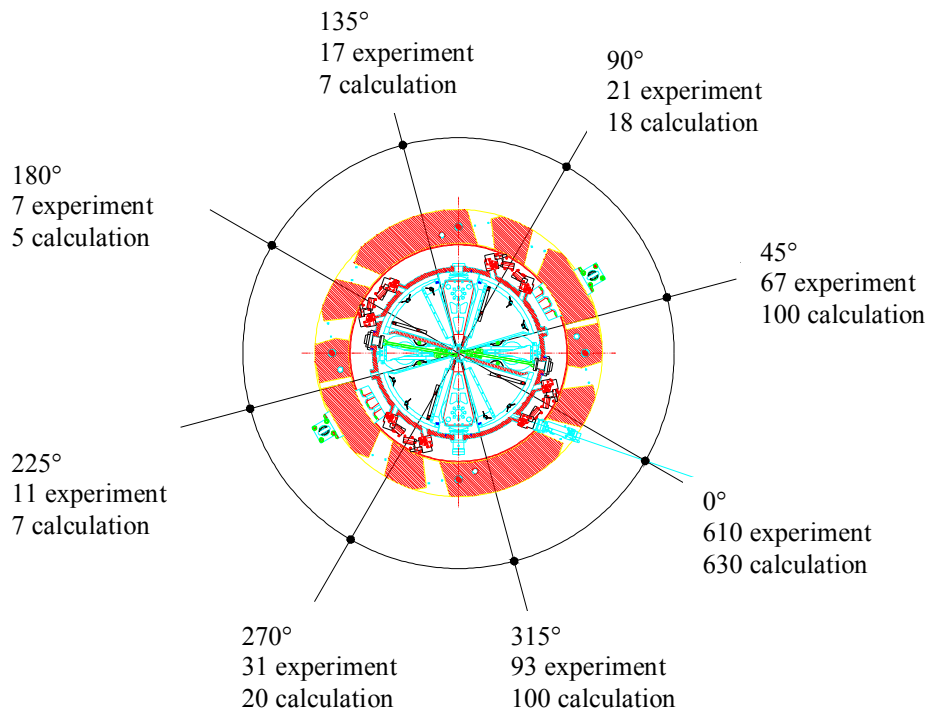


Fig.1 The distribution of neutron dose equivalent around the Cyclone 18/9

Cyclotron vault with maze was calculated by Monte Carlo code MCNP4B. In the input file for the vault simulation was used Cyclone 18/9 as a neutron source, which was calculated in the previous steps. Design of the vault with maze was optimized for various position of the target for  $^{18}\text{F}$  production as is shown on Fig.2.

Neutron dose equivalent rates per 100 mA of proton beam at various points of cyclotron vault with maze (see Fig.2) are given on Tab.6.

Tab.6 Dose equivalent rate at detection points

Direction of proton beam	Detection point	Dose equivalent rate in $\mu\text{Sv h}^{-1}$ per 100 $\mu\text{A}$
A	802	0,52
A or B	806	0,32
A	807	1,4
B	808	1,7

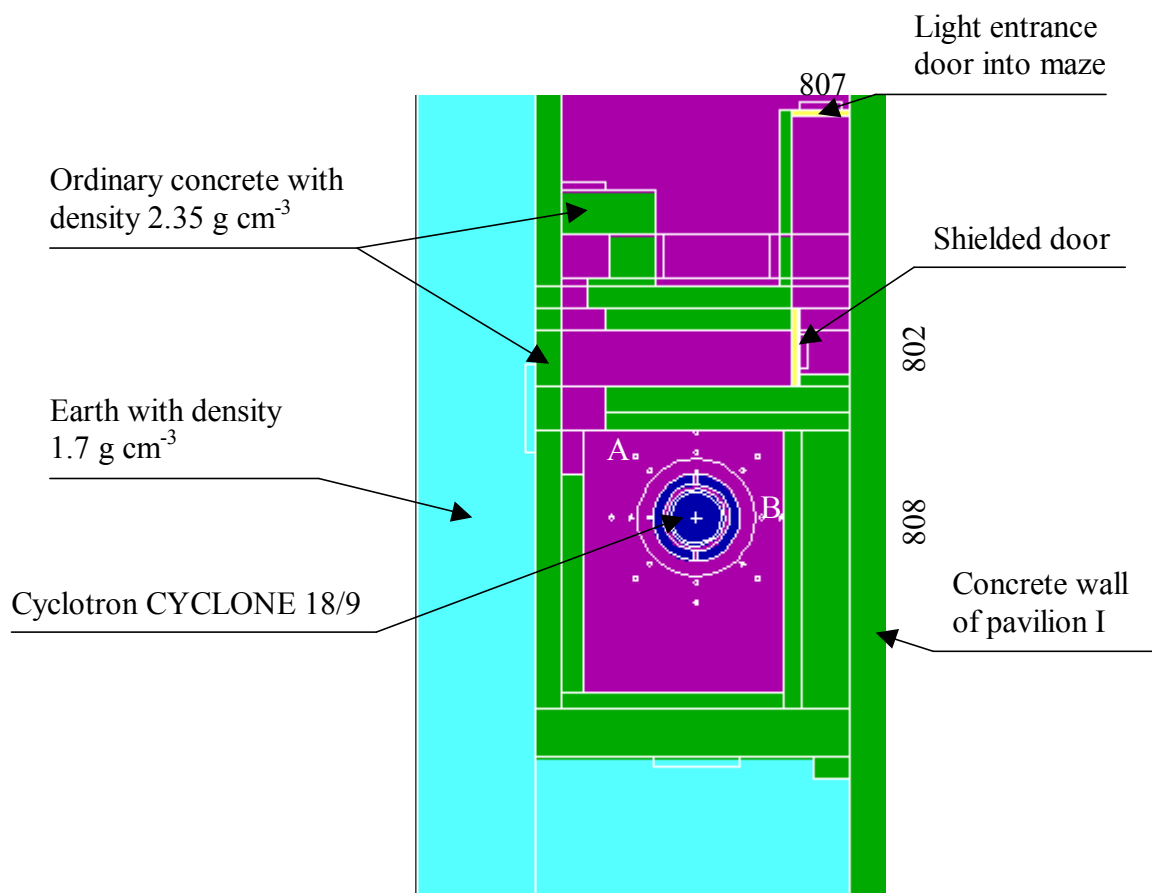


Fig.2 Layout of cyclotron vault with maze. Localities 802, 807 and 808 are detection points behind the shielding

### Conclusion

It was proved that design of vault of cyclotron (main source of ionizing radiation in pavilion I) can be effectively and precisely solved by calculations based on Monte Carlo simulation of the cyclotron as a source of ionizing radiation.

This procedure is also useful for determination of gamma radiation from induced radioactivity in various parts of cyclotron and for assumption of radioactivity in concrete of vault walls.

# EXPERIENCE WITH AIRBORNE DETECTION OF RADIOACTIVE POLLUTION (ENMOS, IRIS)

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## 1. Fast and Reliable Radiation Detection

Technological advancements of our society create with the increased level of comfort, increased risk of either unintentional or intentional radioactive pollution. New instrumentation and processing techniques can rapidly produce visual images of areas exposed to radiation. Protecting the health of the population in case of a nuclear accident is an essential social priority.

Monitoring of existing levels of natural and manmade radioactive contamination, in and around nuclear installations and nuclear materials handling facilities is a valuable reference in case of a nuclear accident.

Fast deployment of airborne radiation monitoring systems in the case of nuclear accidents is essential. The portability of the new range of instrumentation with accurate navigation, data acquisition and real time processing can provide fast and low cost estimates of potential problems.

There are two principle ways how to produce radiation images of selected areas.

<b>Ground data collection</b>	<b>Airborne data collection</b>
On foot, or ground vehicle	Helicopter or Fixed wing aircraft
<b>Advantages:</b> Very detailed image depending only on the selected grid size or speed of the data collection (vehicle). Different measurements such as Alpha, Beta and Gamma can be obtained.	<b>Advantages:</b> Fast, no access limitation. Almost instant classification of the area. Efficient and large coverage per time unit. Integral view of the area, with resolution of the measurements related to the altitude and speed of the aircraft. Low cost per surveyed line km. Geographically consistent data. With proper safety procedures, limited potential to dangerous exposure.
<b>Disadvantages:</b> Limited or restricted access (buildings, terrain). Time consuming, takes long time to cover suitable area. Geographically non-consistent data. Dangerous in high radiation levels.	<b>Disadvantages:</b> Initial cost of renting/using aircraft or helicopter. Somewhat limited spatial resolution. Only Gamma measurements can be recorded.

When comparing the total cost, time required and consistency of collected data, it is apparent that an airborne data collection is a practical answer even for very high spatial resolution requirements. With the advanced GPS navigation systems, it is possible to collect information flying at the altitude of 50 to 80 meters with the flight line separation of 50 meters or less.

It would take approximately two working days to collect data over an area of 10 by 10 kilometers at an altitude of 80 meters with line separation of 80m. This represents in total 1,200 km of direct flight lines (discounting the turns). Preliminary maps can be processed within hours after the flight and final images of the area can be ready in a couple of days. To acquire similar volume of data on the ground would take a considerably longer time at a much greater cost. Changing meteorological conditions over the time of survey would negatively influence the quality of collected data.

To interpret collected data on radioactivity properly, the data must be recalculated into absolute (physical) units. Real time absolute measurements are needed in case of an emergency response, to activate emergency procedures. In post-flight processing more sophisticated algorithms are used to generate very accurate images of the measured area.

## 2. Required Instrument Sensitivities

The detector sensitivity is proportional to the exposure and its exposed area. For non-emergency airborne foot printing, a larger size of detectors is desirable. The NaI(Tl) crystal (10x10x40cm) is a generally accepted detector size. An array of these detectors (usually 4) are mounted in a quad or dual detector package forming a detection surface area of 40x40cm = 1600cm<sup>2</sup>.

Statistical character of the measurement relates the sensitivity related to the unit of time (usually 1 second).

Resolution of the detector limits the detectability of the neighboring energy peaks.

Air to ground model is quite complex. The statistical nature of the detection process further complicates this model.

The altitude relation for <sup>40</sup>K (1.4612MeV) measurement, with the detector size of 40x40x10cm, expressed in estimated number of detected Gamma particles for one Bq (homogenously saturated layer with the density of 1.66 g/cm<sup>3</sup>, Lambertian distribution) is:

Altitude	Gamma/Bq	Bq/Gamma
30m	0.4121	2.430
60m	0.3221	3.104
90m	0.2333	4.286
120m	0.1621	6.169
150m	0.0965	10.36

Statistically, the results vary within +/- 3 standard deviations.

Calculated sensitivities of one detector for individual radioactive elements are approximately:

Element	<sup>137</sup> Cs	<sup>60</sup> Co	<sup>40</sup> K	<sup>214</sup> Bi	<sup>208</sup> Tl
Units	<b>Bq/m<sup>2</sup></b>	<b>Bq/m<sup>2</sup></b>	<b>Bq/kg</b>	<b>Bq/kg</b>	<b>Bq/kg</b>
Normal Background (area related)	7000	1000	250	30	40
Detection sensitivity +/- 50% @90m	6200	3320	214	112	50

We can conclude that the above discussed detector package (minimum of two detectors), with the exception of  $^{60}\text{Co}$ , and  $^{214}\text{Bi}$ , can quite reliably measure levels of the background radiation and anomalies exceeding twice the background levels. The lower energy secondary peak improves the  $^{214}\text{Bi}$  measurement sensitivity.

### 3. Comparisons of Ground and Airborne Measurements

The evaluation of fall-out and soil contamination after the Chernobyl accident a nation-wide survey was organized in the Czech Republic between June 26 and 28, 1996. Altogether 1,300 bare soil samples were collected on sites not shielded by buildings, shrubs and trees, with no grass surface, preferably on agricultural land but not on sandy soil, not tilled since April 26, 1986, with the slope less than  $3^\circ$ . The deposition ranged in more than three orders of magnitude. The arithmetic mean of  $^{137}\text{Cs}$  deposition calculated from the lognormal distribution is 6.5 kBq/m<sup>2</sup>.

Results of various methods and equipment for the assessment of ground contamination were used and compared during the airborne survey in 1996. Reasonable agreement of the results proved that all described methods are ready for contamination mapping carried out by Radiation Monitoring Network in case of a nuclear accident.

Results of the survey in some measurement points

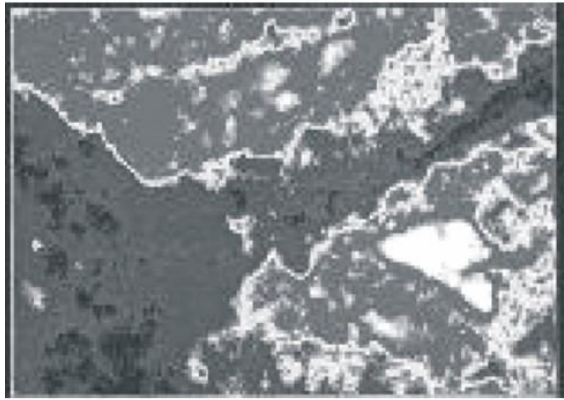
sampling or measurement point	$^{137}\text{Cs}$ deposition [kBq/m <sup>2</sup> ] (July 1996)		
	Airborne measurement ENMOS	Ground in-situ gamma spectrometry	Soil sampling
1	12.7	15.4	20.9
2	29.3		19.9
3	25.1	20.3	23.1
4	40.1		80.9
5	28.0	35	
6	35.7	43.1	
7	32.7	37.6	
8	37.2		51
9	64.4		44.4
10	63.6	52.7	
11	64.4	72	
12	59.0	61.4	
13	43.4	29.6	
14	41.7	30.9	
15	28.3	13.4	23.2
16	57.7	62.4	76.4
17	50.2	30.9	
18	55.1	43.4	
19			7

The highest deposited levels of  $^{137}\text{Cs}$  found were 120 kBq/m<sup>2</sup> (corrected for the radioactive decay to the time of the first nation-wide survey in June, 1986).

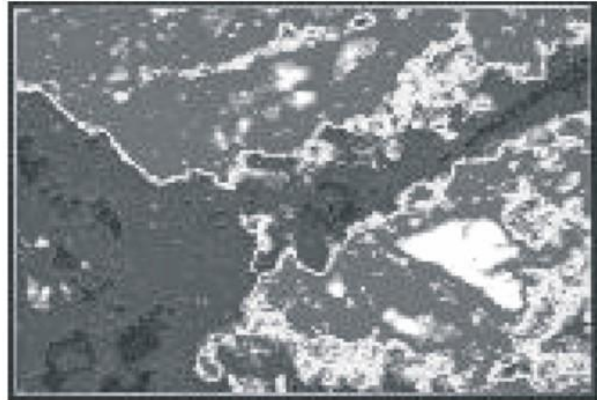
#### 4. Post flight - data enhancement techniques.

Proprietary **PRAGA** (Pico Radiation Air to Ground Algorithm) processing is using spectral deconvolution technique. It is based on minimizing the residual value by “turning on/off” present or expected to be present radio nuclides.

Data from one (4-liter) detector was processed by the **PRAGA**. Data from array of 8 (32 liters) detectors was processed by the standard window stripping technique.



One 4x4x16 inch detector (4 litre volume) with full **PRAGA** Th channel processing.



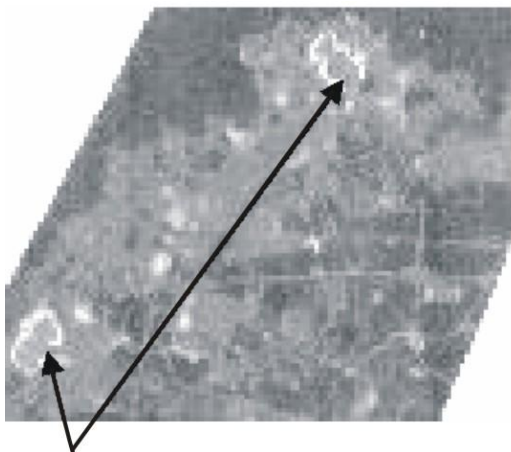
Eight 4x4x16 inch detector (32 litre volume) with conventional Th window processing.

There are other processing techniques such as the **Singular Value Decomposition (SVD)** developed by J. Hovgaard.

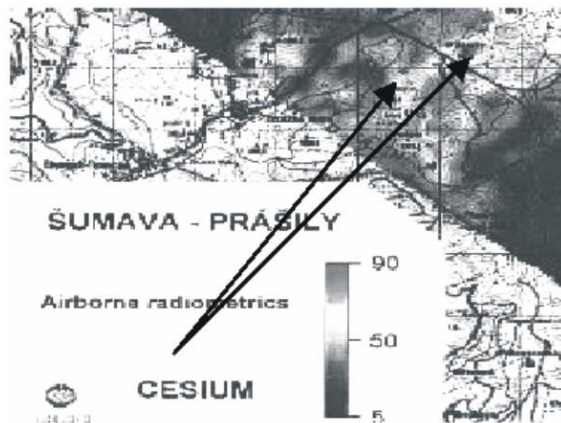
#### 5. Examples of airborne Radiation detection

Ontario Hydro – Bruce Nuclear Power Plant, Canada.

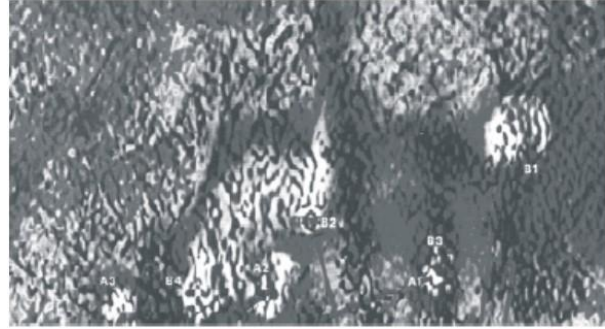
The total dosage of low energy radiation up to 3MeV was calculated by the **PRAGA**. Investigating the Cosmic channel (above 3MeV) it was found that there is additional radiation caused by the  $^{16}\text{N}$  (Nitrogen). This element is mostly detected above the reactors



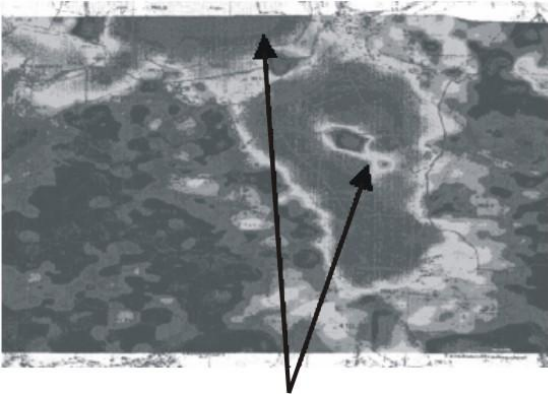
High intensities in Cosmic channel at the nuclear power plant.



Airborne detection of  $^{137}\text{Cs}$  after Chernobyl accident at Zelezná Ruda in Czech Republic (measured in  $\text{Bq}/\text{m}^2$ )



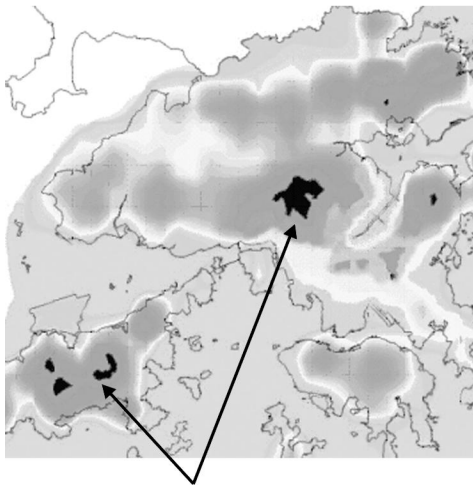
$^{60}\text{Co}$  and maps of Maralinga, Australia 40 years after nuclear tests



Germany – open pit  
Uranium contamination from mining.

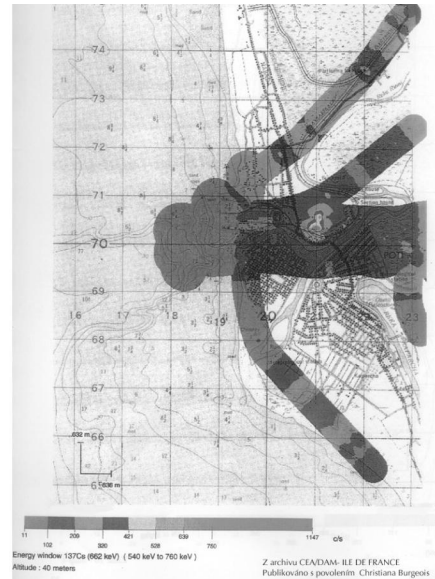


Australia tailings contaminating river



Altitude problems

Hong Kong Observatory. Hilly terrain, wather and highdensity of tall buildings make themeasurement of real radiationcontamination extremely difficult.



Searching for lost nuclear materials in Georgia



### **Notes on provided graphics:**

All of the maps and pictures are converted from color. The gray representation of colors is not always accurate. Should you be interested in receiving digital images, please contact the authors.

### **6. New instrumentation.**

New, almost portable instrumentation allows the user to see selected data in real time and immediately after the flight. Information can be transmitted to the control center.

Complete precise navigation is integrated into system supporting UTM and ADRG together with geographical database and color – high brightness touch screen.

### **7. Conclusions**

Many examples of real situations assessed on the basis of data gained by the airborne measurements demonstrate, that the use of airborne data is reliable, fast and relatively inexpensive. Short period of time required for data acquisition assures data consistency. Practically unrestricted access provides good and homogeneous data.

Today advanced measuring and processing techniques are result of many years of hard and slow progress mostly in airborne geophysics, together with advancements in mathematics, physics, data processing and electronics.

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### **9. Acknowledgements**

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# CHANGES OF ATMOSPHERIC $^{14}\text{C}$ CONCENTRATIONS IN TWO LOCALITIES OF SLOVAKIA IN THE LAST FIFTEEN YEARS

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## **Introduction**

The natural level of the  $^{14}\text{C}$  in the atmosphere has been changed mainly by the influence of the nuclear bomb tests realized in beginning of sixtieth years of the last century. Approximately in the year 1963, the  $^{14}\text{C}$  concentration from nuclear testing reached the peak value of about 100% above the natural level in the atmosphere of the northern hemisphere [1]. After the nuclear moratorium for the atmospheric tests, the  $^{14}\text{C}$  concentration in the atmosphere has decreased because of the exchange processes between the atmosphere and the other carbon reservoirs [2-6]. The decrease of  $^{14}\text{C}$  in the atmosphere is reasonable influenced also by the dilution of the atmosphere with inactive carbon, produced by combustion of the fossil fuel. The influence of the fossil fuel  $\text{CO}_2$  emissions on  $^{14}\text{C}$  concentration in the atmosphere can be observed and studied also on the regional level [5-7].

In this contribution the results of the long-term measurements of the atmospheric  $^{14}\text{CO}_2$  in Bratislava and at Zlkovce stations are presented and the influence of the anthropogenic  $\text{CO}_2$  emissions on the  $^{14}\text{C}$  concentration is discussed.

The samples taken in Bratislava station (48° 9' N, 17° 7' E, 164 m a. s. l.) with 0,5 million inhabitants correspond to highly industrialised region. Zlkovce (48° 29' N, 17° 40' E, 162 m a. s. l.) is situated approximately 60 km NE from Bratislava in a flat agricultural area. The nearest pollution source that can influence the  $^{14}\text{C}$  concentration in the atmosphere is the nuclear power plant Jaslovske Bohunice, approximately 5 km WNW from Zlkovce.

## Methods

For the carbon isotope measurements in the atmosphere the monthly large-volume samples of atmospheric CO<sub>2</sub> have been continuously collected at a height of 15 m above the ground surface by the dynamic absorption of CO<sub>2</sub> in NaOH solution [8]. Further CH<sub>4</sub> was prepared from the sample for filling the low-level proportional counter, which was used for the counting of the <sup>14</sup>C decays [9].

A few ml of CO<sub>2</sub> liberated from the sample in form of BaCO<sub>3</sub> were analysed using a mass spectrometer for the determination of the isotopic ratio of <sup>13</sup>C/<sup>12</sup>C. The δ<sup>14</sup>C values were calculated relative to NBS oxalic acid activity and their standard deviations are on the level of ± 6 ‰. Results are presented as Δ<sup>14</sup>C values and they were obtained from δ<sup>14</sup>C values by the correction on the isotopic fractionation.

## Results and discussion

In Bratislava the <sup>14</sup>C activity has been measured in atmospheric CO<sub>2</sub> since 1967 [10] but simultaneously in both localities the <sup>14</sup>C measurements have been carried out since 1987.

In the Fig.1 the courses of the annual mean Δ<sup>14</sup>C measured in Bratislava and at Zlkovce are shown. For a comparison, the dashed line in Fig.1 shows the long-term trend of annual mean of Δ<sup>14</sup>C in background air over Europe. Since 1987 to 1993 the annual mean

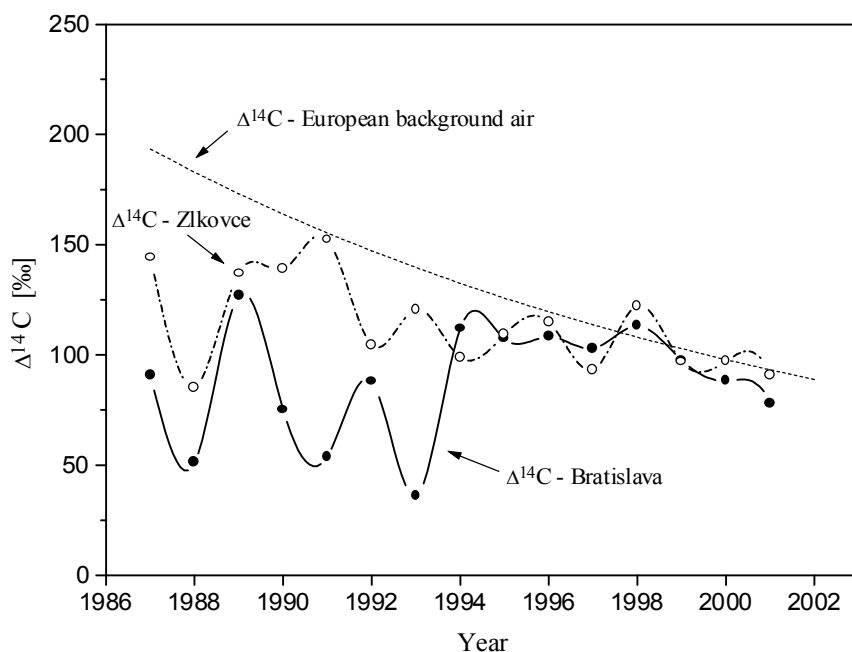


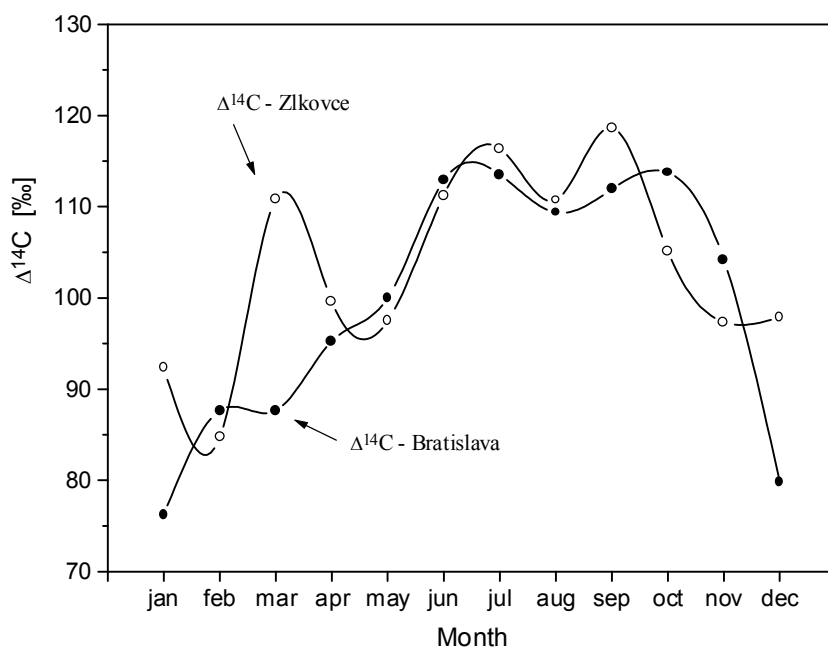
Fig. 1. The annual mean values of the Δ<sup>14</sup>C in the atmospheric CO<sub>2</sub> in Bratislava and at Zlkovce. The dashed line represents the long-term of the annual mean Δ<sup>14</sup>C in the background air over Europe [6, 11].

Δ<sup>14</sup>C values are significantly below the Δ<sup>14</sup>C values in background air for both stations. However, in the last years for both sites the measured Δ<sup>14</sup>C move much closer to the European background Δ<sup>14</sup>C trend line.

Before 1994 [11] the annual mean values of  $\Delta^{14}\text{C}$  in the atmospheric  $\text{CO}_2$  collected in Bratislava were on average about 50 % lower and at Zlkovce about 20 % in comparison with the  $\Delta^{14}\text{C}$  background level. At that time  $\Delta^{14}\text{C}$  at Zlkovce were about 50 ‰ higher than in Bratislava. In this period in Bratislava, very low and sometimes even negative monthly mean  $\Delta^{14}\text{C}$  values were measured mainly in January and February evidently as a consequence of the high input of  $^{14}\text{C}$  free fossil fuel  $\text{CO}_2$  into the atmosphere (Suess effect). The relative high increases of  $\Delta^{14}\text{C}$  in spring and early summer months were caused rather by the depletion of the atmospheric  $^{14}\text{C}$  in winter months than by the injection of the stratospheric air into the troposphere. According to the expectation, the Suess minima of  $\Delta^{14}\text{C}$  were not as distinct at Zlkovce as they were in Bratislava, although they were also identified. But in some months also the high  $\Delta^{14}\text{C}$  reaching the value up to 300 ‰ were measured (in March 1989 in both localities and in April, May and July 1991 only at Zlkovce) [11]. These  $\Delta^{14}\text{C}$  excesses could be of the technogenic origin.

Since 1994 there were no longer such marked differences in annual mean  $\Delta^{14}\text{C}$  values between the two stations. This could be explained by the decreasing of the fossil fuel  $\text{CO}_2$  emissions in Slovakia after 1990 and their stabilization after 1994 [12, 13].

For the stable period 1995 – 2001, also the average annual courses of  $\Delta^{14}\text{C}$  are presented in Fig.2 for both localities. The courses are similar each other and reach the maxima in summer months (August) and the minima in winter (from December to February), with



seasonal variations from 76 to 113 ‰ for Bratislava and from 85 to 118 ‰ for Zlkovce. The high value ( $\Delta^{14}\text{C} = 111$  ‰) in average annual  $\Delta^{14}\text{C}$  course for Zlkovce in March is due to the high measured monthly mean  $\Delta^{14}\text{C}$  value in March 1998 at this station.

Fig. 2. The average annual courses of the  $\Delta^{14}\text{C}$  in the atmospheric  $\text{CO}_2$  in Bratislava and at Zlkovce. Both courses were obtained as the mean of the  $\Delta^{14}\text{C}$  data of years 1995 – 2001.

The set of the regularly monthly measured  $\Delta^{14}\text{C}$  enabled to study also the long-term trend of the amplitudes of the  $\Delta^{14}\text{C}$  annual courses. The results of the harmonic analysis are shown in Fig. 3.

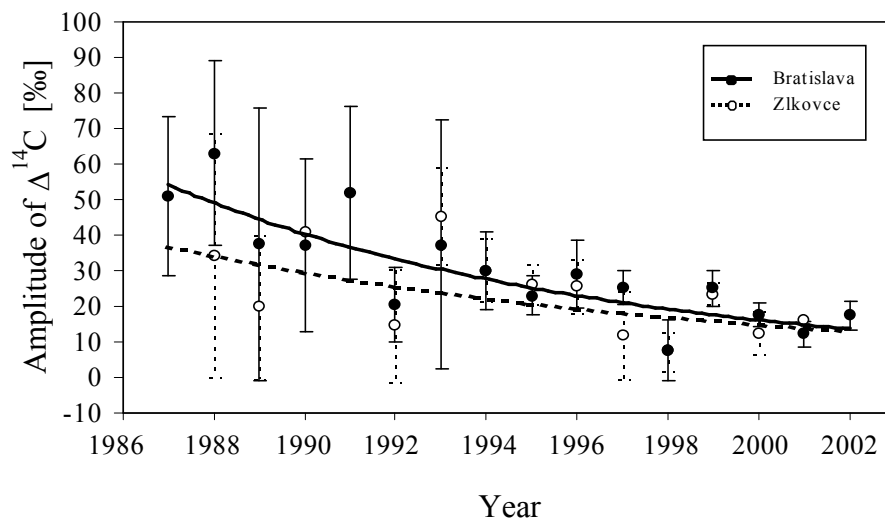


Fig. 3. The amplitudes of the annual courses of the  $\Delta^{14}\text{C}$  in the atmospheric  $\text{CO}_2$  in Bratislava and at Zlkovce. The trend lines were calculated according to the equation  $y = 3,22 \cdot 10^5 \cdot e^{-0,0997 \cdot x}$  for Bratislava and  $y = 1,012 \cdot 10^5 \cdot e^{-0,0886 \cdot x}$  for Zlkovce. In these equations  $y$  is the annual mean of  $\Delta^{14}\text{C}$  and  $x$  is the calendar year.

The amplitudes of the annual  $\Delta^{14}\text{C}$  courses present the decreasing trend. While in 1987 the amplitude of the annual course of  $\Delta^{14}\text{C}$  in Bratislava was equal to 55 ‰, in 2001 was equal to 14 ‰ only. In the atmosphere of Zlkovce the amplitudes of the annual  $\Delta^{14}\text{C}$  courses are lower than in the atmosphere of Bratislava, but they decrease also. This effect is probably due to the decrease of the fossil fuel  $\text{CO}_2$  emissions into the atmosphere in winter months.

## Conclusion

A high variability of the annual mean  $\Delta^{14}\text{C}$  in the atmospheric  $\text{CO}_2$  was observed at two not very distant stations until 1993. In this period the annual mean values of the  $\Delta^{14}\text{C}$  in heavily polluted atmosphere of Bratislava were about 50 % lower and at Zlkovce about 20 % lower compared to  $\Delta^{14}\text{C}$  in European background air. After 1993 the annual mean  $\Delta^{14}\text{C}$  was in both sites close each other and in 2001 reached the value approximately 8 % only above the  $\Delta^{14}\text{C}$  natural level.

The  $\Delta^{14}\text{C}$  values in the atmospheric  $\text{CO}_2$  present the annual variations. The amplitudes of the annual courses are decreasing with time and in 2001 they reached the value only 14 ‰.

The observed  $\Delta^{14}\text{C}$  behavior in the atmosphere provides an unique evidence of the decrease of fossil fuel  $\text{CO}_2$  emissions into the atmosphere.

### Acknowledgements

This study was funded by the Scientific Grant Agency of the Ministry of Education of the Slovak Republic (VEGA projects No. 1/4194/97, 1/7653/20, 1/0246/03) and by the International Atomic Energy Agency, Vienna (Res. Contract No. 9093/RO).

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# **PARTICIPATION OF THE RADIATION HYGIENE LABORATORIES TO THE WHO/UNEP GLOBAL ENVIRONMENTAL RADIATION NETWORK**

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## **Short history**

In December 1987, a WHO-UNEP meeting held at SCPRI (Service Central de protection contre Les Rayonnements Ionisantes - Le Vesinet, France) set up the basis of the international network GERMON (Global Environmental Radiation Monitoring Network) as an extension of existing network "Global Environment Monitoring Systems" (GEMS). The accident from Chernobyl certainly was the important nuclear event influencing this decision. The aim of the GERMON network is to initiate programmes for the routine monitoring of the environmental radioactivity and to ensure a quick interchange of credible data in case of major accidental radioactive releases, as well as the development of intervention devices in the member states running such programmes. The responsibility of the Co-ordinating Collaborating Centre (CCC) has been given to the French Service Central de Protection Contre les Rayonnements Ionisants (SCPRI). In 1994, this Service became the Office de Protection Contre les Rayonnements Ionisants (OPRI).

On May 28<sup>th</sup> to 30<sup>th</sup> 1990, the representatives of 5 WHO regions, of the CCC and of the WHO secretariat held a meeting in Suzdal to establish the working system of the network (structure, methodology and communication dynamics). Consequently, by the end of 1990, the first 9 member states: Canada, Czechoslovakia, Ethiopia, India, Korea, Romania, Sweden, United States of America, and Yugoslavia, transmitted their results. The number of the participants has increased to 15 in 1991, and to 44 in 1998. A new meeting of the executives of the GERMON network was held on April 27<sup>th</sup> to 30<sup>th</sup> 1992 in Montgomery (USA), but no significant changes in the working system were considered necessary.

**The participation of the Radiation Hygiene Laboratories to the WHO/ UNEP Global Environmental Radiation Monitoring Network ( GERMON)**

The Ministry of Health has a national network consisting of 23 radiation hygiene laboratories; 19 of these are included in the framework of county divisions of public health, and the other 4 are compartments of the regional institutes of public health. WHO designated the Institute of Public Health from Bucharest as **National Contact Centre**, in charge with communicating the results obtained by the national laboratories on the indicators of environmental radioactivity, according to the established methodologies. The main indicators considered are: ambient gamma dose, radioactivity of the air, of the precipitation, and of the milk.

Following the measurement and transmission protocols of the CCC, the Radiation Hygiene Laboratory from the Institute of Public Health has established a methodology to be followed by the laboratories of the national network:

1. For the determination of the ambient gamma dose, the quantity required to be measured is the absorbed dose rate in air (nGy/h) at 1m above the ground; the measurements is carried out weekly with a dosimeter having a detection limit around 20 nGy/h.
2. Air activity: For the average weekly gross alpha and beta activity of the aerosols (Bq/m<sup>3</sup>), the air is continuously sampled on a membrane filter, for a period of 18 hours, once a week (on Monday or Tuesday), with a low flow rate of 20-30 l/min. After two days, the filter will be measured with a measurement chain having a maximum background of 1 cpm for gross alpha activity, and 10 cpm for gross beta activity. In order to determine the presence or absence of the I-131 and Cs-137, the filter should be kept and measured again 15 days later by means of a gamma spectrometer with a NaI(Tl) or Ge(Li) detector.
3. For milk (Bq/l) and deposition (rain, snow, dry deposition) (Bq/m<sup>2</sup>): the activity is measured directly, in a Marinelli box (appr. 1 L), with a gamma spectrometric chain, with a detector of at least 5 cm diameter.

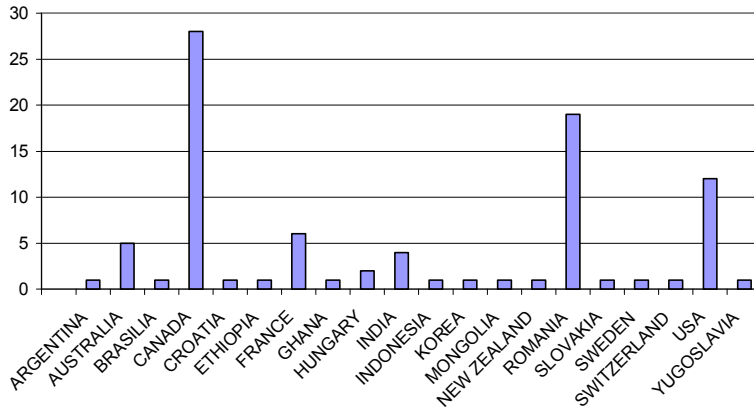
Since the above are minimal requirements, additional accurate measurements can be carried out (and reported), such as: specific radioactivity (K-40, Cs-137), and radiochemical analysis for Sr-90.

The data should be reported to the IPH - Bucharest on the form sheet, within the first 15 days of the month following a quarter.



## Results

Figure 1 shows the countries participating in this program as for 1993, and the number of monitored locations.

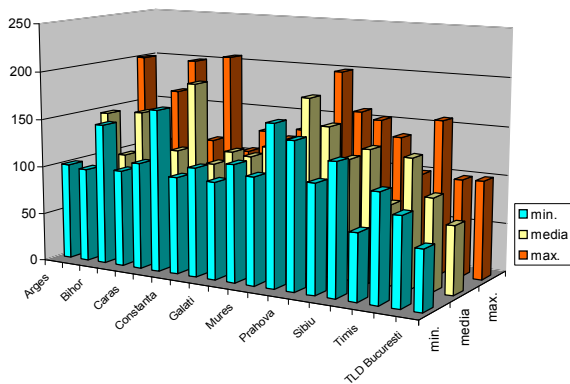


**Figure 1:**  
The participating countries and number of the sampling locations for each participating country

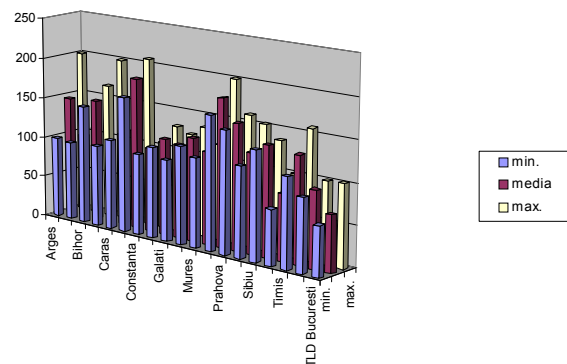
Romania participates in this program from the very beginning (1990) through the network of radiation hygiene laboratories of the MH.

Apart from the importance of participating in such an international programme, another important achievement is that the monitoring covers the entire country.

The ambient gamma dose is one of the indicators followed by all the laboratories, although some of them do not yet have instruments of the desired sensitivity. The long term determinations carried on by each laboratory represent, in the absence of a major nuclear event, the baseline values ("the local zero") (Figure 2), which are extremely useful in case of radiological emergencies.



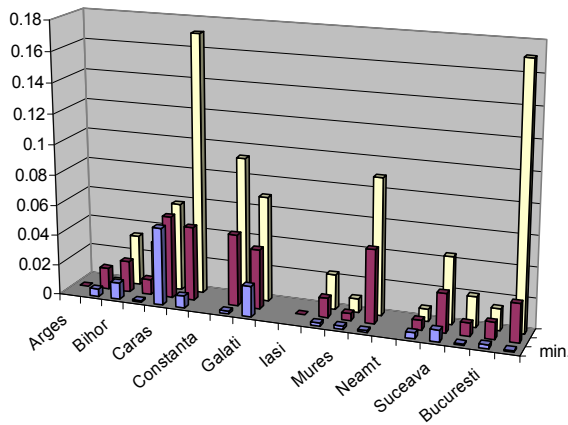
**Figure 2:** Regional ambient gamma dose rate (nGy/h)



**Figure 3:** Regional ambient gamma dose rate (nGy/h)

The CCC organised an intercomparison exercise for this quantity, using integrative detectors (TLD). Figure 3 shows the results of this comparison. Most of the results showed lower values for this type of detectors than for the measurements carried out by means of dosimeters.

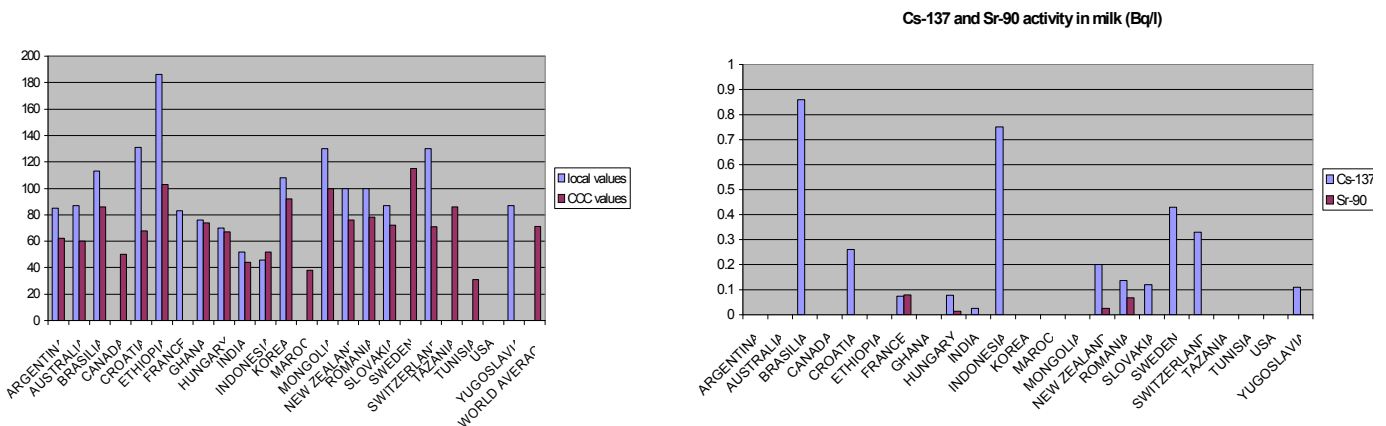
Another monitored indicator is the aerosol activity, measured 3 days after sampling. Figure 4 shows the data from the Romanian laboratories.



**Figure 4:** Regional gross beta activity of aerosols (Bq/mc)

For milk radioactivity, the Romanian radiation hygiene laboratories monitored all the parameters that characterise both natural, and artificial radioactivity. The Figures 5 to 6 present our results as compared to those of other participating countries. The values presented for Romania, are a 5-year average for the entire country, and the results for the other countries are the figures reported in 1993.

**Figure 5,6:** Results of the intercomparison exercise



# **RODOS SYSTEM IN THE SLOVAK REPUBLIC – ITS IMPLEMENTATION AND ADAPTATION**

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## **Introduction**

Experience gained after the Chernobyl accident clearly demonstrated the importance of improving administrative, organisational and technical emergency management arrangements in most of the European countries. The tragic events in the US in September 2001 also obliged to pay more attention to emergency management.

Significant progress has been achieved in the information technological (IT) and methodological areas of emergency management and response through national initiatives and with support of the EC under its 4th and 5th Framework Programmes. Many tools are now available for operational use and their application in practice, such as the installation of gamma monitoring systems in East and West European countries, the collection and evaluation of their data under EURDEP, the further development and installation of the decision support system RODOS for (pre-) operational use in many East and West European countries, the realisation of the prototype data and information exchange system, and the development of tools and methods for exchanging data and information between neighbouring countries. Most of these new methods and IT tools can be equally applied everywhere in Europe and thus contribute to a more and more common and coherent approach for off-site emergency management and response in Europe.

## **Key futures of RODOS**

RODOS - **R**ead-time **O**n-line **D**ecision **S**upport system for multi-user operation in national/regional emergency centres responsible for off-site nuclear emergency management. RODOS provides continuously updated comprehensive, consistent and timely information as

input to decision-making in local/national/ regional /European scales, in the early and later phases of an accident on all types of emergency actions and countermeasures.

Information processing in RODOS is performed in four levels and is following:

LEVEL 0 ⇒ OSY (operating subsystem):

acquisition, storage and checking of data from radiological monitoring networks and measurements and their presentation together with geographical information,

LEVEL1 ⇒ ASY (analysing subsystem):

continuously updated diagnosis and prognosis of the radiological situation using meteorological data and forecasts, source term predictions, and monitoring data,

LEVEL 2 ⇒ CSY (countermeasure subsystem):

estimation of the extent and duration of countermeasures together with their radiological and economic consequences,

LEVEL 3 ⇒ ESY (evaluating subsystem):

evaluation and ranking of countermeasure options by balancing their respective benefits and disadvantages including practicability, acceptance and socio-political aspects.

### **Implementation of the RODOS in the Slovak Republic**

With support of the European Commission's (EC) ECHO programme "Implementation of the RODOS Decision Support System for Off-Site Emergency Preparation and response in the Emergency centres in Poland and the Slovak Republic", RODOS system version 3.13 was implemented in the Slovak Republic with main objective to accelerate the implementation of the system within national emergency preparedness arrangements [1].

Within the ECHO project, the National Centre of RODOS was established at the Nuclear Regulatory Authority of the Slovak Republic (NRA SR). VUJE was established as technical support organization and interactive user. As a data supplier and passive user were established the Slovak Centre for Radiation Monitoring, Slovak Hydro-Meteorological Institute (SHMI), Jaslovske Bohunice and Mochovce NPP's.

Lessons learnt from the RODOS installation in the Slovak Republic and similar projects in other countries are following:

select a well-experienced HP provider for installing and configuring the hardware and system software firewall during installation and test phase (FZK access),

early start of customisation (collection of site and country specific data, conversion to appropriate formats),  
established communication lines to the data providers and transfer of data (real-time meteorological data and forecasts),  
fast communication lines to remote users (>128kB/s),  
qualified personnel available (UNIX administrator, experience with format conversion and real-time data),  
intensive training in RODOS operation,  
full engagement of the ERC staff during installation and operation.

### **Adaptation to local-national conditions and integration of the RODOS system to the existing administrative emergency management structures**

After implementation of the RODOS system the adaptation to local conditions has continued. Connecting to the ECHO project the national project “Further development and preparation of the RODOS integration to the emergency planning and management in the Slovak Republic” supported by the government of the Slovak Republic was established. Within this project the following main tasks were performed or are in progress:

creation and updating of RODOS databases,  
adaptation of the RODOS system to the on-line data sources,  
comparison of the RODOS modules for assessment of the accident situation with models developed and operationally used in the Slovak Republic,  
integration of the guidelines for the RODOS system to the existing emergency procedures,  
training programme and plan for the integration of the RODOS system to the emergency planning and crisis management.

The important part of the project is also participation in the international exercises (DSSNET, DEKO) of the users and R&D community.

During this phase of RODOS integration to the existing administrative emergency management structures version 4.0F of RODOS with several patches and significant updates have been installed and tested. New version 5.0 of RODOS was installed in April 2003 and used for the 3<sup>rd</sup> DSSNET exercise preparation and performance as a concerted action of many institutes (UJD, VUJE, Slovak Weather Service, FZK) [2]. All relevant Slovak geographical

data and statistical data on grids were adopted and implemented into the RODOS Geographical Information System (RoGIS). New site and plant characteristics data for both Jaslovske Bohunice and Mochovce were implemented into the RODOS Fix-Database. The database of source terms for reactor unit types VVER 440/V213 and VVER 440/V230 was created, tested and fully integrated to the RODOS system. Update of the databases for RODOS modules FDMT, LCMT, EMERSIM and EARLYCONS with taking into account new Slovak legislation was completed. Specification, adaptation and design of the meteorological on-line data transfer to the Real-time database of the RODOS system has been completed. Within this part of the project the development of the ALADIN to RODOS interface has been completed and implemented. Verification of the dispersion modules ATSTEP and MATCH and their comparison with Real-time Accident Release Consequence (RTARC) system was completed. Update of the emergency procedures and their integration to the national emergency arrangements was completed. The guidelines for preparation scenarios, organizing and evaluation of emergency exercises with RODOS system was developed by VUJE and adopted by UJD SR. The training courses for RODOS users and operators were prepared and conducted by VUJE for participants of Nuclear Regulatory Authority of the Slovak Republic, NPP Jaslovske Bohunice, NPP Mochovce, Slovak Centre for Radiation Monitoring, Slovak Hydro-Meteorological Institute, Slovak Army, organizations from Ministries of Health, of Interior, of Environment, of Economy and of Soil Management.

Lessons learnt from the (pre-) operational use of RODOS in national emergency centres are following:

Transition phase (at least up to two years after installation):

- completion of network connections to data providers and remote users,
- customisation of RODOS to local and national conditions;

Maintenance and support phase:

- support in case of problems with hardware, software and network connections (telephone and E-mail hotline with FZK),
- extension of users and databases, new system software or hardware components, new RODOS versions/patches,
- realisation of user requests, support in emergency exercises, cyclic training of operators and users.

The RODOS system has been used during three international exercises, conducted by FZK (Germany), within the DSSNET project under 5<sup>th</sup> Framework Programme of European Union aiming at facilitating the communication between the users and R&D community. RODOS was also used as a main tool during the Regional Emergency Exercise „DEKO 2001“, conducted by NRA SR in cooperation with VUJE, where Hungarian, Polish and Slovak National RODOS centres, KFKI (Hungary), OSSKI (Hungary), IAE (Poland) and VUJE (Slovak Republic) took part. Testing of the National RODOS centres of three participating countries took place [3].

### **Conclusions**

As RODOS could become a standard tool for emergency preparedness in the EU and CEE region lessons learned within projects mentioned in the paper were valuable contribution to other similar international and national projects. Effective working arrangements and links have been established between some institutes having competence in various aspects of emergency preparedness and response in about 20 European countries, with Forschungszentrum Karlsruhe as main responsible institute for system development and DSSNET project co-ordination. This network contributes to the more cost-effective use of resources for the further improvement of decision support for nuclear emergencies. More importantly, it will enhance trust and confidence between people, which is essential for responding effectively and coherently to any future nuclear accident that may have implications on a European scale.

### **References**

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- [2] T. Duranova, Preparation, scenario, structure and performance of the 3<sup>rd</sup> DNNET exercise. 4<sup>th</sup> Meeting of the DSSNET members, Krakow, Poland, 3 and 4 July 2003.
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# **OPERATIONAL INTERVENTION LEVELS AND RELATED REQUIREMENTS ON RADIATION MONITORING DURING PRE-RELEASE / RELEASE PHASE OF AN ACCIDENT**

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Any radiological monitoring in case of nuclear accident or in case of any other event with possible radiological consequences should be strongly related to real and applicable operational intervention levels. In case of emergency there is not sense to measure quantities which cannot be readily and clearly compared with pre-determined intervention levels.

Required outputs of emergency radiological monitoring in various phases of an accident and rationale of these requirements is discussed.

1. At the beginning of an accident the most important output of monitoring is the knowledge of the source term (release prognosis and especially the estimation of real release to the environment). These data are inevitable in the process of implementation of disaster measures like evacuation, sheltering, taking iodine tablets, measures to reduce inhalation of radioactive substances.
2. Time Integral of Air Volume Activity should be used for evacuation, sheltering, taking iodine tablets, measures to reduce inhalation of radioactive substances. Another measures which can be applied on the base of this output of emergency monitoring is avoiding of staying outdoors, change into different clothing after staying outdoors, measures for storage of contaminated clothing, measures on exchange filters in buildings and trucks (cars). In the field of agriculture should be considered immediate harvesting, closing of greenhouses, closing of inflow into cisterns.
3. Further output of monitoring is surface activity on the terrain. This can serve for implementation of evacuation, sheltering, relocation, resettlement and avoiding of hard work or sport. In the field of agriculture immediate harvesting, closing of greenhouses, closing of



inflow into cisterns should be taken in mind. This parameter can be very important for decisions about cultivation of plants which are not intended for consumption, change of crop rotation, forestation in the later phase.

4. Air volume activity can be used in the field of agriculture for consideration about immediate harvesting, closing of greenhouses and closing of inflow into cisterns.

5. At the boundary of contaminated areas is important to check contamination of vehicles. This parameter is used for making decisions like decontamination or turning vehicles back.

6. Activity on filters is another important parameter. On the base of its value measures like exchange of filters, protection measures on exchanging and storage of filters are applied.

7. Activity in agricultural products and foodstuffs has strong impact on measures relating leafy vegetables, green fodder, pasture, milk, meat and fruits.

8. For limitation of stay and avoiding of watering there is necessary to know volume activity in water or sediment.

9. Similarly mass activity in waste is necessary for limitation of staying and protective measures during collection and transport of waste.

10. Dose rate from deposit on the terrain can be used for implementation of evacuation after the passage of cloud.

In various phases of an accident various intervention levels are important and consequently various radiological quantities should be preferably measured. Distinguished tasks or aims of monitoring in different phases of accident have strong influence on methods of monitoring, instrumentation and capabilities of monitoring groups.

Required tasks and outputs of monitoring:

1. Estimation of expected release to the environment in pre-release phase (estimation of expected source term).

Note: it is expected that there is estimated possible release (in Bq) of noble gasses, iodines, caesiums, tellures. The part of prognosis should be information whether early or late release is expected. There should be reported basic information on initiating event (information like: release from the reactor core, or other specific information).

2. Estimation of real release in Bq or in Bq per time interval in release phase.

Note: knowledge of estimated real release of noble gasses, iodines, caesiums and tellures is expected. The part of this information should be the time from the end of chain reaction. The part of report should be information that (eventually) there is not release (release is interrupted or seems to be finished). The part of report should be information on initiating event of the accident or information about the state of reactor core damage (if appropriate).

3. Measurement of time integral of air volume activity in  $\text{Bq}\cdot\text{h}/\text{m}^3$  in release/post-release phase.

Note: the highest priority is to evaluate time integral of air volume activity of I-131, especially in villages or cities, and eventually in the area of NPP. At second, there is need to know time integral of air volume activity of Cs-137, Te-132. Report should contain information on coordinates of the place of measurement (from GPS) and basic information about rain intensity, too (like: without rain, heavy rain, weak rain).

4. Measurement of air volume activity in  $\text{Bq}/\text{m}^3$  in release/post-release phase.

Note: there is need to know air volume activity of I-131, Cs-137. This part of report should contain information on coordinates of the place of measurement (from GPS). Basic information on rain intensity is demanded, too (information like: without rain, drizzle, heavy rain).

5. Measurement of surface activity on the terrain in  $\text{Bq}/\text{m}^2$  after the cloud passage and in post-release phase.

Note: at first there is demanded knowledge of activity of I-131, Cs-137, Te-132. At second knowledge of activity of Sr-90 or Am-241, Pu-239 will be required and applied. The part of report should be information on coordinates of the place of measurement (from GPS).

6. Measurement of dose rate from deposit on the terrain in  $\text{Sv}/\text{h}$  after the cloud passage and in the post-release phase.

Note: the most important is measurement of dose rate from deposit in those places where evacuation should be performed before the cloud passage but has not been realized up to know, and in the area of NPP. At second, this measurement should be focused on detection of hot spots in villages and in the vicinity of villages in order to apply protection measures. This measurement technique must enable to distinguish between dose rate from deposit and dose rate from the activity in air. The part of report should be information on coordinates of the place of measurement (from GPS).

7. Measurement and evaluation of contamination of vehicles in  $\text{Bq}/\text{m}^2$ .

Note: there is demanded knowledge of surface activity of Cs-137, I-131, Te-132. The part of report should be information on coordinates of the place of measurement (from GPS).

8. Measurement and evaluation of activity on the filters (in the buildings and in vehicles), in  $\text{Bq}$ , in post-release phase.

Note: there is required knowledge of activity of I-131, Te-132, Cs-137 on the filters. The part of report should be information on coordinates of the place where the filter was collected (taken from the car) and/or measured (from GPS).

9. Measurement and evaluation of mass activity in agricultural products and foodstuffs, in Bq/kg, in post-release phase.

Note: at first there is required knowledge of mass activity of Cs-137, I-131, Sr-90. At second complete measurement of radionuclide contamination of the sample can be required. The part of report should be information on coordinates of the place where the sample was collected (from GPS).

10. Measurement and evaluation of mass activity in the waste, in Bq/kg, in post-release phase (sewage plants, waste dumps, contaminated waste).

Note: there is demanded mass activity of the waste, especially activity of Cs-137, I-131 and Te-132. The part of report should be information on coordinates of the place where the sample of waste was collected and/or measured (from GPS).

Compatibility of emergency monitoring is crucial requirement. In case of emergency a lot of results is expected to be collected and appropriate evaluation and correct application of these results can hardly be done without high compatibility regarding calibration, units, geographical coordinates systems, etc.

# **RADIATION MONITORING NETWORK OF THE SLOVAK HYDROMETEOROLOGICAL INSTITUTE**

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## **1. Monitoring Network of the Slovak Hydrometeorological Institute**

### **1.1 History**

The extensive development of peace using nuclear energy after the world war II and the tests of nuclear weapons in the 50ies caused the remarkable increasing of artificial radioactivity in the atmosphere. Therefore many hygienic and meteorological services have started to monitor radiation.

In 1962 the department „Radiation of atmosphere“ has been established under the Slovak Hydrometeorological Institute in Bratislava. Artificial beta radiation of atmospheric deposition has been measured in the selected meteorological stations from 1962 to 1991. Within 1962, 1963, after the testing of nuclear weapons in the 50ies and the beginning of the 60ies, the maximum values were reached in the former Czechoslovakia. Increased values were recorded again in 1968-1971, 1974, 1981 and in 1986 after the Cernobyl accident. In 1991 the measurmets of dose rate started.

### **1.2 Monitoring of dose rate**

At present SHMI operates in its monitoring network 23 detectors GammaTracer fy Genitron, one mobile detector and one stanby detector. All activ detectors are placed in the professional meteorological stations in the selected parts of Slovakia. First one of these detectors was installed in 1999 and they replaced former type of detector (FAG). Last two detectors were installed in 2002. Detector GammaTracer has range of measurement from 20 nSv/h to 1 Sv/h.

The detectors are calibrated every 2 years in the Slovak Institute of Metrology in compliance with the calibration plan. Last calibration was very successful, all detectors showed good parameters.

**Radioactivity Monitoring Network of SHMI  
(31. 12. 2002)**

<b>N.</b>	<b>Ident.</b>	<b>Station</b>	<b>Long.</b>	<b>Lat.</b>	<b>m a.s.l.</b>
1.	11812	Malý Javorník	48 15	17 09	584
2.	11813	Bratislava-Koliba	48 10	17 06	340
3.	11819	Jaslovské Bohunice	48 29	17 40	176
4.	11826	Piešťany	48 32	17 50	163
5.	11841	Žilina - D. Hričov	49 14	18 37	310
6.	11855	Nitra	48 17	18 08	135
7.	11856	Mochovce	48 17	18 27	261
8.	11858	Hurbanovo	47 52	18 12	115
9.	11867	Prievidza	48 46	18 36	259
10.	11880	Dudince	48 10	18 52	140
11.	11903	Sliač	48 39	19 09	314
12.	11916	Chopok	48 59	19 36	2008
13.	11918	Liesek	49 22	19 41	692
14.	11927	Lučenec	48 20	19 44	214
15.	11930	Lomnický štít	49 12	20 13	2635
16.	11933	Štrbské Pleso	49 07	20 05	1355
17.	11938	Telgárt	48 51	20 11	901
18.	11952	Poprad-Gánovce	49 02	20 19	695
19.	11958	Kojšovská Hoľa	48 47	20 59	1242
20.	11968	Košice-letisko	48 40	21 13	231
21.	11976	Stropkov	49 13	21 39	216
22.	11978	Milhostov-Trebišov	48 40	21 44	105
23.	11993	Kamenica nad Cirochou	48 56	22 00	117

### 1.3 Aerosol monitors

SHMI operates 4 aerosol monitors in Hurbanovo, Lucenec, Stropkov and Liesek. Filters from these monitors are analysed in the Institute for Health (Cs-137, Be-7).

On the base of bilateral agreement between the Austrian Ministry of Agriculture, Forestry, Environment and Water-Management and the Slovak Ministry of Environment Austrian side gave into the ownership of the Slovak side an automatic aerosol monitor AMS-02 including container and weather station. This monitor was installed in meteorological stations Jaslovske Bohunice on 4-th October 2001. The Slovak Ministry of Environment provides the Austrian Ministry of Agriculture, Forestry, Environment and Water-Management with the readings of this monitor, free of charge, for at least 3 years and vice versa, the Austrian side gives the readings of the Austrian aerosol monitors to the Slovak Ministry of Environment free of charge. At present national monitoring center in Bratislava-Koliba is connected via ISDN line with Jaslovske Bohunice and Austrian center providing the data exchange.

## **2. Database of Radiation Monitoring**

### **2.1 Collecting of data**

Radiation data (dose rate in the unit nSv/h) are collected via the Institute network to the MSS (message switch system) in the meteorological station Bratislava-airport. The service program FTP-Watch runs on the server RADMON in SHMI and every 10 minutes the data from MSS are inserted into the database. The 2hour and 24hour averages are computed on the server automatically.

### **2.2 Database**

Two backuped servers work in the system of radiation monitoring under Windows 2000 Server operating system and MS SQL Server database system.

Database contains one table for radiation data and several tables for configurations, catalogues of stations and additional tables.

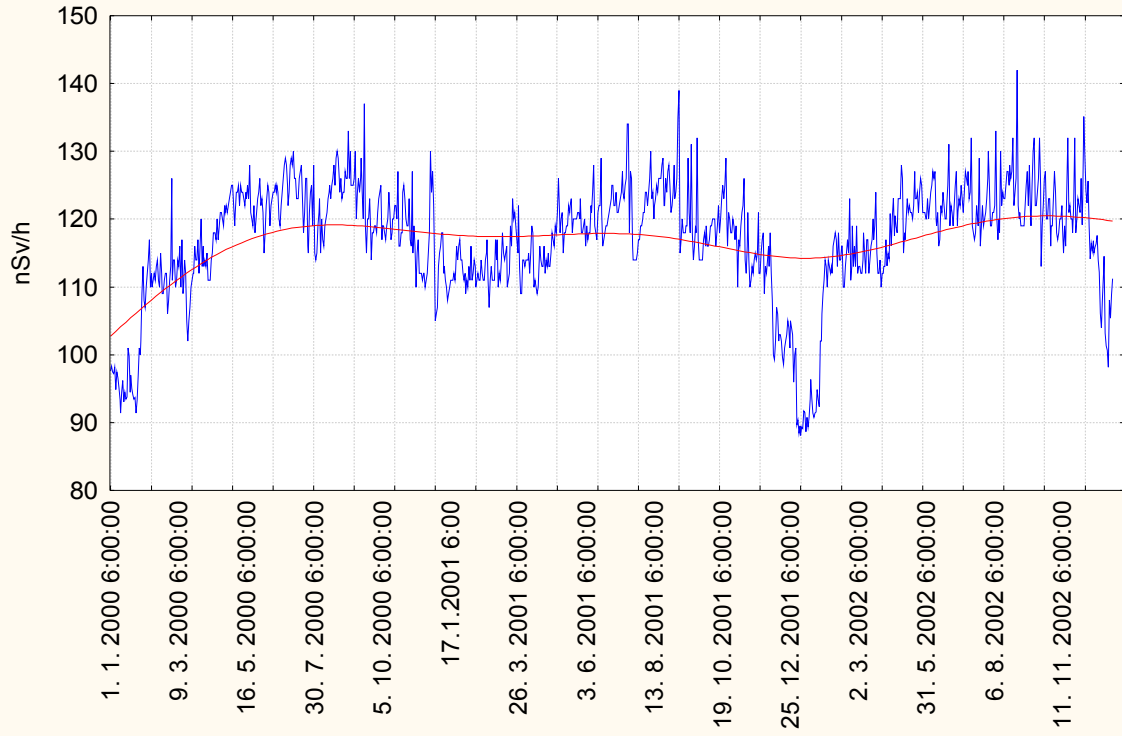
Database works in environment client-server. On client PC runs the user front-end application. This application provides to display the data using many filters, to display tables with configurations concerning technical equipment, to display maps, graphs, etc. There is the possibility to store data into the archives, to make reports.

This extensive database gives good opportunity to mathematical and statistical analysis. DTS (Data Transformation Services) as one part of SQL Server gives possibility to design reports in many formats based on SQL scripts.

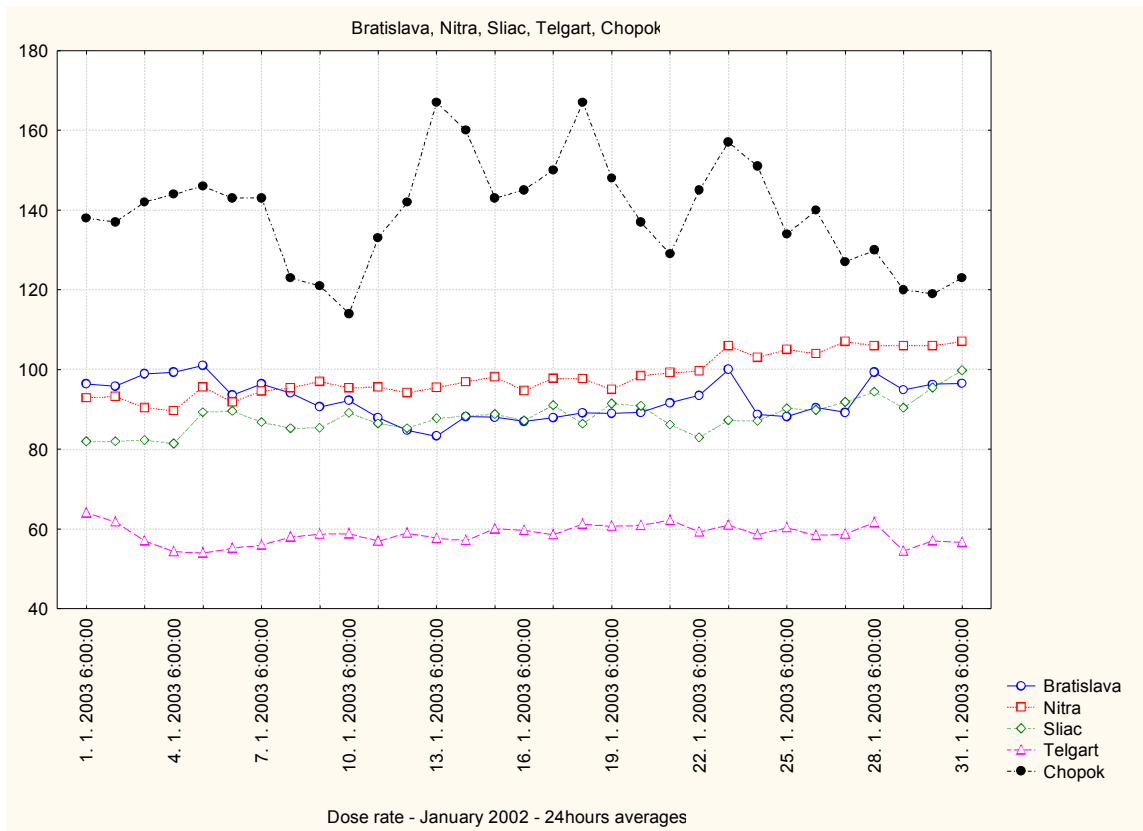
### **2.3 Data Analysis 2002, 2003**

Time series from monitoring points are analysed by the environment of the statistical software STATISTICA 6.0. and presented in reports. Some results are enclosed.

# JASLOVSKE BOHUNICE 2000 - 2002



(Dose rate, 24-hour averages)



### 3. Cooperation in the Data Exchange on the national level

In the frame of Unit database of radiation data in the Slovak Republic, SHMI cooperates with other partners like: Slovak Army, Civil Defence, Ministry of Health, Slovak Power Plants. At present bilateral data exchange with Slovak Army is running and with other partner is prepared.

### 4. International Data Exchange

#### 4.1 European Commission Joint Research Centre Ispra

SHMI cooperates with European Commission Joint Research Centre (EC JRC) in Ispra in the frame EURDEP (European Union Data Exchange Platform). At present we use in the data exchange with EC JRC new version of format EURDEP 2.0. We send data from our monitoring network on ftp server of SHMI every 24 hour and then the data are downloaded to database in Ispra. We have access to the data on the EC JRC server via internet on the base of username and password.

#### 4.2 Austria

Data between SHMI and Radiation Warning Centre Vienna are exchanging by means of directories on the radiation monitoring server of SHMI. Every 10 minutes data from 336



Austrian stations are stored into the directory on our server and then inserted into the radiation database. Every 10 minutes data from our monitoring network are stored to the directory on server on our side and then downloaded to the Austrian side.

EURDEP format version 1.3 is used.

### **4.3 Hungary**

On the base of agreement between Hungarian Ministry of Environment, Hungarian Ministry of Interior and the Slovak Ministry of Environment, SHMI started the data exchange with Hungary Meteoservices in summer 2002. Leased line Bratislava – Budapest of capacity 16 kbit/s was established. Data files with the radiation data in the EURDEP 2.0 format are exported from our database every 10 minutes and then files are downloaded to the server in Meteoservice Hungary. Files with radiation data are downloaded from Hungarian side each 1 hour (10 minutes averages).

Data between SHMI and Meteoservices Hungary and SHMI and Radiation Warning Centre Vienna are transmitted via Regional Meteorological Data Communication Network.

# Clean-up Levels for Recovery of a $^{137}\text{Cs}$ Contaminated Site in the Slovak Republic

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The 19 km long banks of the Bohunice NPP waste water recipient (Manivier canal (0.3 m<sup>3</sup>/s) and Dudváh River (0.8 m<sup>3</sup>/s)) has been identified as contaminated by  $^{137}\text{Cs}$  as a result of two accidents on the CO<sub>2</sub> cooled and heavy water moderated NPP-A1 unit in 1976 and 1977. Until 1992, NPP waste water had been derived through a 5 km-long canal to the Dudváh River (Q<sub>a</sub>=1.8 m<sup>3</sup>/s) conjuncting with the Váh River (150 m<sup>3</sup>/s) after 13 km downstream at 90 km from Váh's mouth into the Danube River. Between 1976 and 1978, when both accidents happened, construction of a flood control project on Dudváh River had just been being implemented in the length of 8 km upstream of its mouth. In the next upstream part of the River with about 5 km long river section, affected by NPP, the flood control conditions are insufficient and has, hitherto, caused permanent concern of the public.

The contamination of banks and its significance was discovered in 1991 in connection with preparation of a flood control project implementation. As a result, the flood control project implementation was stopped. In 1992 a bank restoration project including shallow burial of removed soil was initiated by the nuclear Power Plant (NPP) who was considered as responsible for the site contamination. A cleanup level of 1 Bq/g  $^{137}\text{Cs}$  was given ad hoc by the authority. The contaminated soil disposal was designed as a subsurface concrete structure with planned capacity 5 000 m<sup>3</sup> inside the NPP area. This area was considered to be as the most acceptable disposal site for the nearby public.

**Radiological characterization of the contaminated banks.** Detailed radiological survey done between 1992 and 1994 shows that the top  $^{137}\text{Cs}$  soil contamination on the banks widely varies from background level to about 20 kBq/kg (3.8 MBq/m<sup>2</sup>) the Dudváh River and reaches 250 kBq/kg for some limited spotty-contaminated sections on the Manivier canal banks. The average  $^{137}\text{Cs}$  activity concentration on the lower part of banks reaches 6.3 kBq/kg

in the top 10 cm soil layer. The overall contaminated area in the site with  $^{137}\text{Cs}$  exceeding 1Bq/g, selected as a preliminary working limit by the authority, has been identified as about 67,000 m<sup>2</sup> and the volume of soil to be removed at this clearance level about 13 000 m<sup>3</sup>.

**Re-consideration of the restoration project.** In view of above new monitoring results, it became clear that the 1 Bq/g of  $^{137}\text{Cs}$  level in soil is too-low for use as justified cleanup criteria. The restoration project showed to be necessary to being re-considered with emphasis on the complexity of proposed restoration project, e.g. large area monitoring in NPP surroundings, analysis of proper bank restoration techniques (removing, clean covering, trenching, fencing) including disposal of the removed soil and siting of designed disposal structure relating to concerned public. As typical feature of these efforts, clear legislation in the field has hitherto absent. There is why a primary demand to develop some principles for evaluation of the justified scale of cleanup measures including appropriate cleanup criteria development showed out to be as of first priority in order to achieve confidence and authorization of the re-considered restoration project. This demand was realized in a tight cooperation with competent hygiene authorities and experts.

Two techniques belonging to more cost-consuming techniques have been selected for contaminated banks cleanup measures: a) the standard removing/disposal of top soil layer for the steep and un-engineered banks and b) mechanical dilution/fixation of contaminated top soil by clean soil cover for the flat contaminated terraces areas. The clean cover technique sufficiently reduces the anticipated radiation risk, however, its price is about 10-time lower comparing to the standard removing/disposal technique.

**Criterial dose assessments and cleanup level developments:** The contaminated banks are accessible for 16,000 residents living in a 3.5 km wide strip alongside the river. Selected exposure pathway scenarios with authorised parameters were applied for characterization dose assessments and development of proper cleanup criteria (see Tab.1). Apart from external exposure pathway, also, ingestion pathways using transfer factors for goats milk and meat (for loamy soil according to the reference [1] ) were also part of the made dose assessments. It was agreed by the authority that exceeding of the derived cleanup levels only justifies implementation of more cost-consuming restoration techniques.

A stay scenario for the evaluation of actual risk on the banks (300 h of fishing in sitting position) and a residential scenario for use of relocated contaminated soil for housing were selected for evaluation of potential risk from the use of contaminated soil. So, according to these conditions, the effective dose from a stay on a bank did not exceed (in 1993) 0.35

mSv/a, although, the potential risk from the use of contaminated soil reached higher levels of effective dose to up to about 2 to 3 mSv/a. The annual collective dose from the stay on the banks was estimated as low, maximally, on the level of about 100 - 200 man mSv, accordingly to not too-intense use of the banks.

Concerning the low collective doses, it was recognized that the benefit from the planned bank restoration is mainly indirect consisting in allowing the implementation of necessary flood control project. In this context, moving of large amount of the contaminated soil from the river banks, and its free release into the environment during a flood control project implementation (and so its possible use for housing) poses the most serious potential risks for the nearby population. Respective cleanup criteria were derived on the basis of the recovery approach of the ICRP [1], accepted by the authority. According to this both the actual dose and potential risk to a critical individuals from the contaminated banks must not exceed 1 mSv/a.

Average  $^{137}\text{Cs}$  contamination level in the top (10 cm) bank soil,  $AL_{50} = 7.1 \text{ Bq/g}$  continuously over 80 m long sections, corresponds to the above critical potential dose constraint requirement. In addition,  $^{137}\text{Cs}$  activity concentrations  $AL_3 = 25 \text{ Bq/g}$  for isolated spots was proposed for the spotty contaminated sections of canal banks.

According to the authorized principles, clean soil cover is sufficient to be applied over 9,500  $\text{m}^2$  of contaminated flat area [3]. On the spotty contaminated canal section, only the isolated spots of contamination is proposed to be removed. So, the resulting volume of soil to be removed from the steep banks and safely buried in a disposal concrete structure inside the Bohunice NPP area equals to about 1,100  $\text{m}^3$ .

The derived criteria are in good relation, also, with the results of the volume distribution of the activity concentration analysis. Cleanup measures, even, for a small part of the contaminated area on the banks (contamination above of  $7 \text{ Bq.g}^{-1}$ ) would lead to significant improvement in remediation of the contaminated banks.

Other crucial aspects influencing the final restoration project were identified as follow: a.) anxiety of the nearby population about the planned subsurface disposal facility for removed soil, b.) limited disposal capacity inside the NPP area and c.) willingness of the Bohunice NPP to pay and support to the bank restoration project implementation.

**Present conditions for implementation of planned bank restoration.** Even at the mentioned wishing political conditions, the developed restoration project has not till been implemented.

It was mainly caused by problems with siting and disagreement of land-cataster owner with placement of the planned disposal facility inside the NPP area.

Meantime, during the past ten year in Slovakia, the social and political conditions have crucially changed (privatization, more democracy, change in finance allocation priorities). Moreover, a general concept for handling with contaminated soil inside the Bohunice NPP area is being now under development within the current decommissioning efforts in the site. It tends to avoid moving and re-disposal of large volume of contaminated soil as it is considered as very expensive. All these changes crucially influence the cost side of the optimization equation for the contaminated bank remediation. Therefore one can expect that the developed cleanup levels including the scale of burial of the removed contaminated soil from the banks will be newly re-considered as soon as the final decision on the necessity of the bank remediation will be made. Probably, the flood control project implementation on the un-engineered part of the Dudváh banks will be the new starting point for addressing again the remediation at least of the concerned contaminated banks.

Tab. 1 Dose factors (DF) and derived inetrvention levels (DIL=1/DF) for selected criterial scenarios relating to 1Bq/g of <sup>137</sup>Cs in soils

Scenario	Geometry factor, g*	t <sub>exp</sub> , h/y	Ingestion pathway**	Ingestion dose rel.unit***	Dose factor DF mSv/y	DIL(1mSv / y ) Bq/g
Stay on banks	0.54 x1.4	300	mi+me	0.4	0.035	28.6
Stay on field	0.67	500	ve+po	1	0.078	12.8
Use of soil 50m3	0.39	1950	ve+po+mi+m e	1.2	0.14	7.1
Use of soil 200m3	0.67	2000	ve+po+mi+m e	1.2	0.21	4.8

\* -  $g = (\text{used dose rate/Bq.g-1}) / (0,118 \text{ mSv.h-1/Bq.g-1})$ , is a geometry factor against a half-definite source,

\*\* - *ve* – vegetable, *po* – potato, *mi* – milk, *me* – meat

\*\*\* - unit for ingestion was chosen as 0,04 mSv/y corresponding to consumption of 110 kg potato, 55 kg root and 55 kg fruit vegetables grown in contaminated soils with 1 Bq/g of <sup>137</sup>Cs and <sup>90</sup>Sr/<sup>137</sup>Cs=0.02

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2. SLÁVIK,O., MORÁVEK,J., Technologies for Environmental Restoration in Slovakia, IAEA-TEC-DOC- 865, Vol.3, May 1996
3. O. Slavik, J. Moravek, M. Vldar, Journal of Radioanalytical and Nuclear Chemistry, Articles, Vol. 209, No 2 (1996), 381-385

# CALCULATION OF THE RADIONUCLIDES CONCENTRATIONS FROM IN-SITU SPECTROMETRY DATA MEASURED BY SEMICONDUCTOR SPECTROMETER

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## **Introduction**

Semiconductor detectors provide very high energy resolution (comparing with scintillation counters) that enables one to distinguish and identify all radionuclides (natural ones as well as man-made contaminants) carrying out in-situ environmental measurements (typically in the reference point 1 meter above ground). While data from the scintillation detector can be processed (and radionuclides concentrations calculated) by deconvolution technique (preparing detection system response matrix for corresponding experimental arrangement), the unacceptably large matrixes should be necessary (to preserve high resolution) for the spectra from semiconductor detectors processing. By this reason, another method of semiconductor spectra processing and calculation of radionuclides concentration in the soil surface layer was designed. The method is based on the two steps:

1. experimental calibration of the detection efficiency angular-energy dependence for individual detector and mathematical fit of this dependence for desired energy interval (usually up to 3 MeV for environmental measurements) and full spatial angle
2. definition and calculation of the *conversion factors* (independent on the detector and detector parameters) that describe (for given experimental arrangement and source energy) relation between the source emission concentration and photon radiance

## **Detection system calibration**

Experimental calibration of the detection efficiency angular-energy dependence for individual detector can be done using set calibration sources covering desired energy interval. Example of such calibration for the real detector (geometry of broad parallel beam perpendicular to the detector enface) is in Fig. 1 that shows also polynomial fit (two continuously joined fits for two energy sub-regions). Fits of relative angular and energy dependence of detection effi-

ciency can be also done from experimental data. Example of such polynomial fit of angular dependence for the same detector and energy line 344.3 keV is in Fig. 2. Fits of angular dependence for all experimental energies and vice versa enable very good description of detection system parameters using cross interpolation.

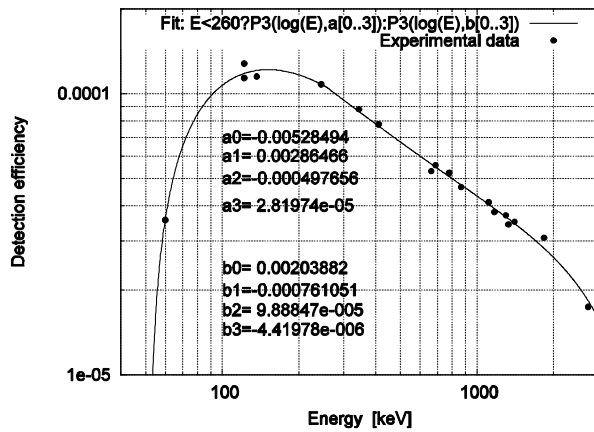


Fig. 1: Example of detection efficiency fit

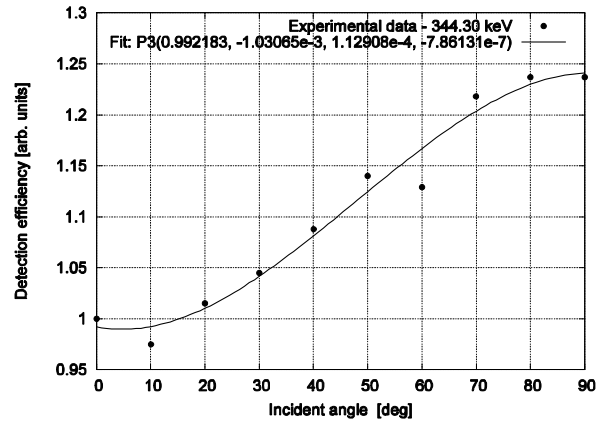


Fig. 2: Example of angular dependence fit

### Calculation of conversion factors

The conversion factors can be obtained (by numerical integration) from photon fluence rate angular-energy distribution, calculated by Monte Carlo model of desired experimental arrangement. Example of such distribution calculated for U-series radionuclides is in Fig. 3, for  $^{137}\text{Cs}$  in Fig. 4. These factors generally depend on soil density (composition, humidity), radionuclides depth distribution model, etc. Taking into account such parameters, we can

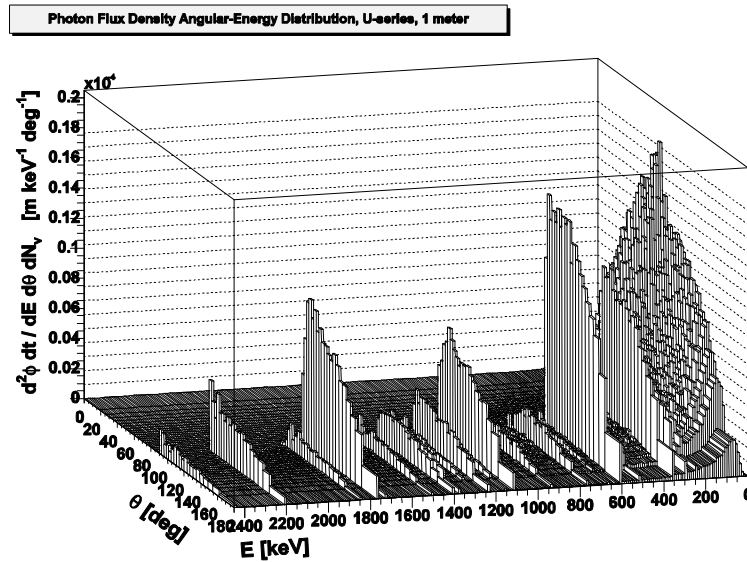


Fig. 3: Calculated photon flux density energy-angular distribution in reference height 1 meter for U-series radionuclides homogeneously dispersed in the saturated soil surface layer (normalised to volume activity  $1 \text{ Bq m}^{-3}$ )

calculate the individual radionuclides concentrations in the soil surface layer (saturated thickness of this layer depends on the source energy) from peak areas of the corresponding source energy lines. Calculated conversion factors for

the most significant energy lines of natural radionuclides (homogeneously dispersed in the soil surface layer) and  $^{137}\text{Cs}$ ,  $^{134}\text{Cs}$  sources (dispersed exponentially in the soil surface layer) are in Figs. 5 and 6. Calculations were done for three soil densities, which enable interpolation for real density if known. Dependence on the soil density is almost negligible for exponential source distribution (see Fig. 6).

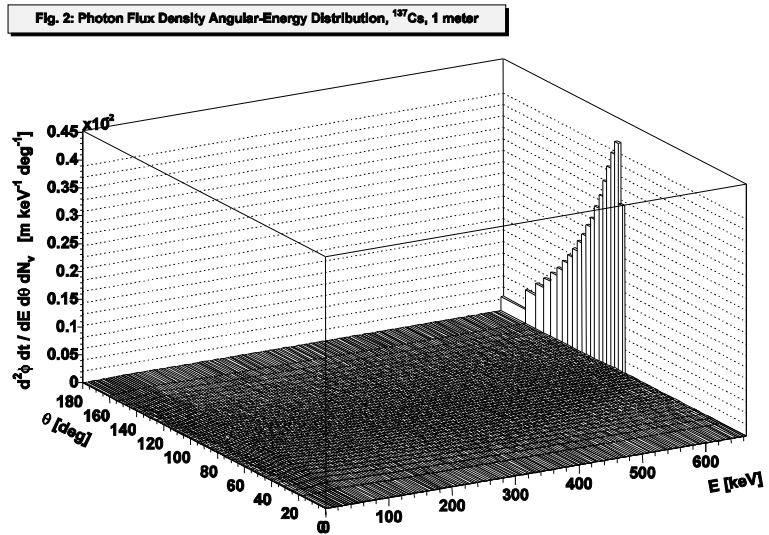


Fig. 4: Calculated photon flux density energy-angular distribution in reference height 1 meter for  $^{137}\text{Cs}$  with exponential depth distribution (relaxation length 3 cm) in the soil surface layer (normalised to effective superficial activity  $1 \text{ Bq m}^{-2}$ )

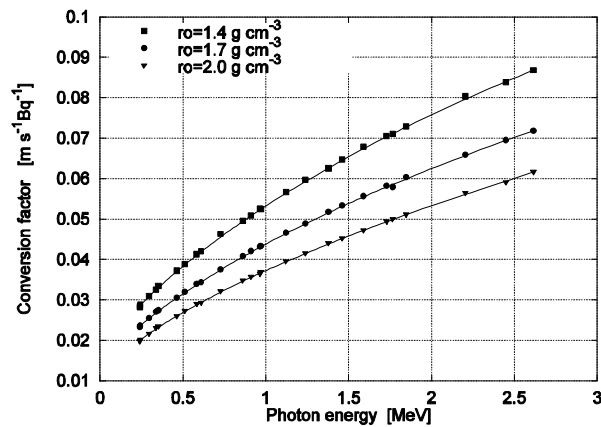


Fig. 5: Calculated conversion factors for most significant energy lines of natural radionuclides (homogeneously dispersed in the soil surface layer), reference height 1 meter and three soil densities

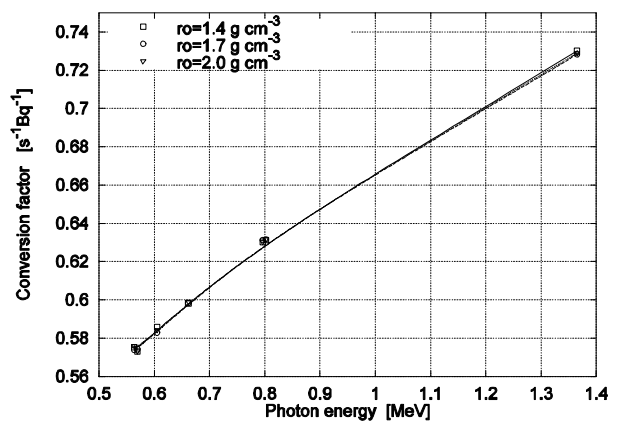


Fig. 6: Calculated conversion factors for the energy lines of  $^{134}\text{Cs}$  and  $^{137}\text{Cs}$  radionuclides dispersed exponentially in the soil surface layer (with relaxation length 3 cm), reference height 1 meter and three soil densities

### Conversion factors for user defined depth distribution models

While homogeneous depth distribution can be considered for the natural radionuclides (U and Th series and  $^{40}\text{K}$ ), the depth distribution of man-made contaminants can depend on many parameters (physical and chemical form, time after deposition, weather conditions, vegetation cover, territory exploitation, etc.). To solve this problem, the conversion factors for the set of



sub-layers of saturated soil surface layer (each set for given energy line) and most significant contaminants were calculated by Monte Carlo simulation. The conversion factors for the user defined depth distribution models can be derived from this data to improve the data processing and interpretation. Example of calculated dependence of conversion factors on mean depth of layer for  $^{137}\text{Cs}$  source and three different soil densities is in Fig. 7.

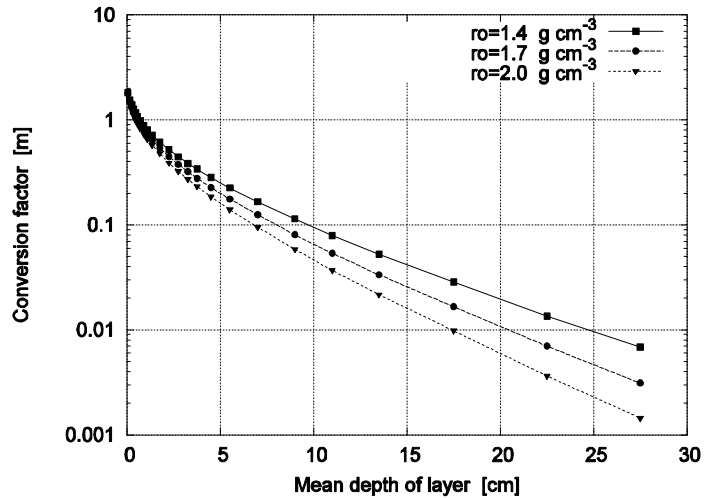


Fig. 7: Calculated dependence of conversion factors on the mean depth of layer,  $^{137}\text{Cs}$  source, reference height 1 meter, three soil densities

Possibility of corrections for the soil density based on the differences of the different photon energies attenuation was studied, but this effect proved to be very small for practical use.

## Conclusion

The computer code based on the described method was designed including calculation of conversion factors for user defined depth distribution models. Inputs of the code are peak areas of the considered radionuclides energy lines (measured in the given arrangement and calculated by any spectra analysis software), known or expected depth distribution models for individual radionuclides (including user defined models) and soil density. The activity concentrations of considered radionuclides and depth distribution models are calculated by the code. Described method was successfully tested and is used for processing of in-situ gamma spectrometry data measured by the spectrometer with semiconductor detector.



# Monitoring of gaseous and liquid releases from NPP Temelín during trial operation

Ing. Jiří Pospíchal, Mgr. Jiří Vokálek, 2003



# THE MAIN MONITORED QUANTITIES

Trial operation - from June 2002 at Unit 1,  
from March 2003 at Unit 2

Inner stacks of both Units - in modes 1-6

Outer stacks of both Units - in mode 6 (Unit outage)

Auxiliary operation building stack - in modes 1-6

**Volume activity of released noble gases**

**Volume activity of released aerosols**

**Volume activity of released iodine nuclides**

**Volume activity of released C-14**

**Volume activity of released H-3**



### BALANCING

#### **Through the medium of authorized conversion factors**

- NG: Ar-41; Kr-85, 85m, 87, 88; Xe-133, 135, 135m, 138
- AE: Cr-51; Mn-54; Co-57, 58, 60; As-76; Nb-95; Zr-95; Mo-99; Ru-103; Sb-124; Te-132; Cs-134, 137
- I: I-131, 133
- LVDIS calculation SW - conversion of released activities to  $E_{(50)} + E$  for the an individual of the critical population group
- Continuously during the year (week, month)

#### **Through the medium of authorized SW, measured Av and concrete meteorological data during the year**

- RD-ETE calculation SW - conversion to  $E_{(50)} + E$  for the an individual of the critical population group including Kr-85
- The definitive balance after year termination



### NOBLE GASES

#### Noble gases monitors

- Spectrometric HpGe detector, measuring volume activity of released air (up to 150 MBq/m<sup>3</sup> Xe-133)
- Recording volume of air released from stacks
- Calibrated for radionuclides within power range of 81–1291 keV



# AEROSOLS

### Sampler with aerosol filter

- Sampling of released air into pressure vessel through aerosol filter (high flow of about 60 m<sup>3</sup>/h, filter size 25 x 20 cm)
- Gamma-spectrometric analysis of samples
- Technological data about released air volume



# IODINE

### Sampler with iodine filters

- Sampling of released air into pressure vessel through pair of filters (low flow about 72 l/min, filters material - impregnated active charcoal)
- Gamma-spectrometric analysis of samples (after homogenization)
- Technological data about released air volume



### C-14

#### Sampler with NaOH solution + $\beta$ scintillation spectrometer

- Absorption of  $\text{CO}_2$  in NaOH solution
- The sampler discerns between organic and inorganic form of C-14
- Creation of sediment during chemical reaction
- Spectrometric analysis of sediment mixed with liquid scintillator





### H-3

#### Molecular screen + $\beta$ scintillation spectrometer

- Absorption of air humidity in molecular screen
- The sampler discerns between organic and inorganic form of H-3
- Desorption of HTO from molecular screen
- Spectrometric analysis of HTO mixed with liquid scintillator



### RESULTS FOR THE YEAR 2002

No limits for released activities (Bq) as for the substances released to the atmosphere

The limits are specified as radiation bodily harm to an individual from the critical population group ( $\mu\text{Sv}$ )

→ 200  $\mu\text{Sv}/\text{year}$  under the Decree

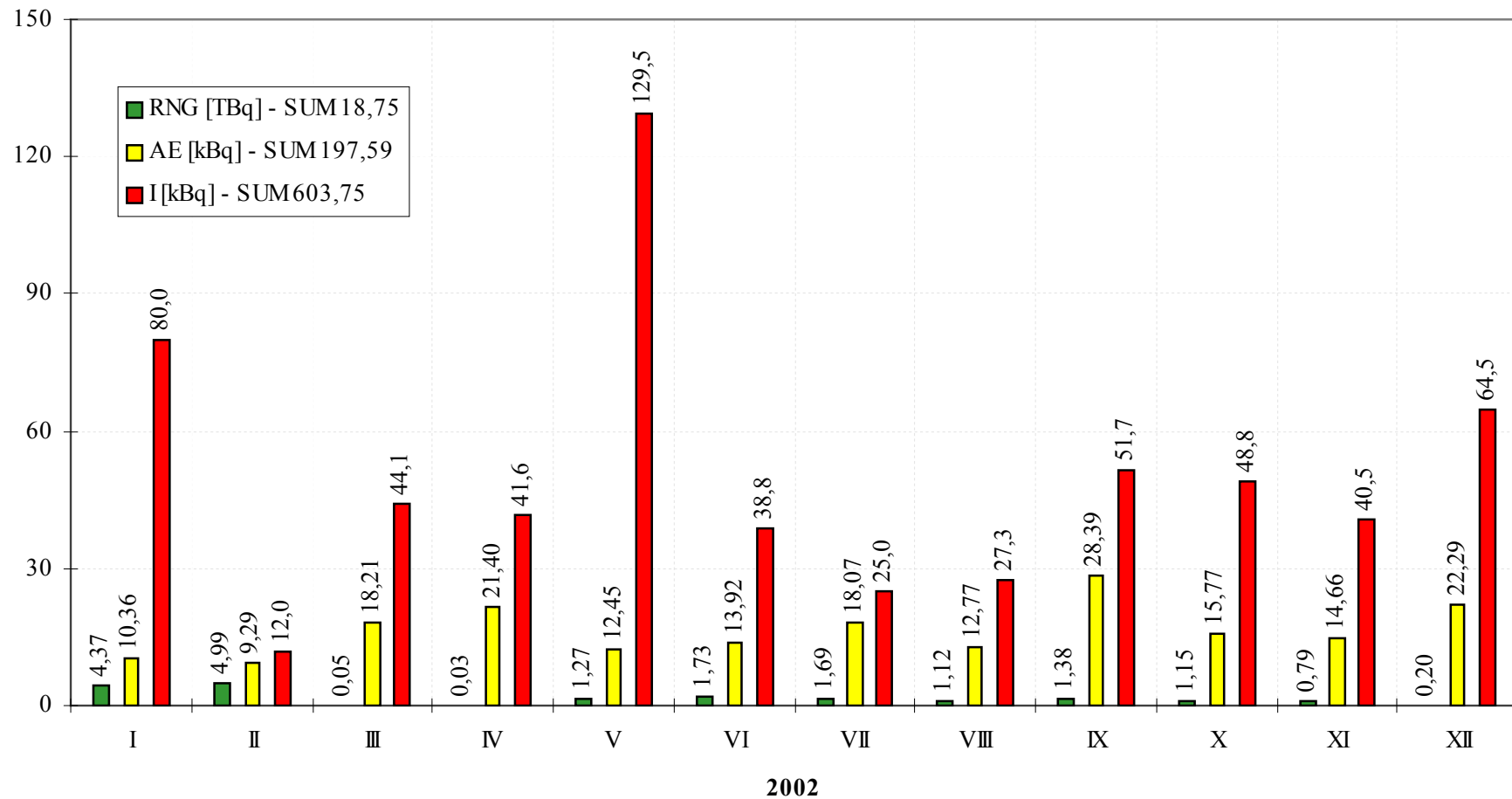
→ 40  $\mu\text{Sv}/\text{year}$  under the Technical Specifications for NPP operation

**The limit of 40  $\mu\text{Sv}$  - exceeded by tenths of per cent, expecting units of per cent in the future** (according to the experience gained at the Dukovany NPP and other NPP's in the world)



## RESULTS FOR THE YEAR 2002

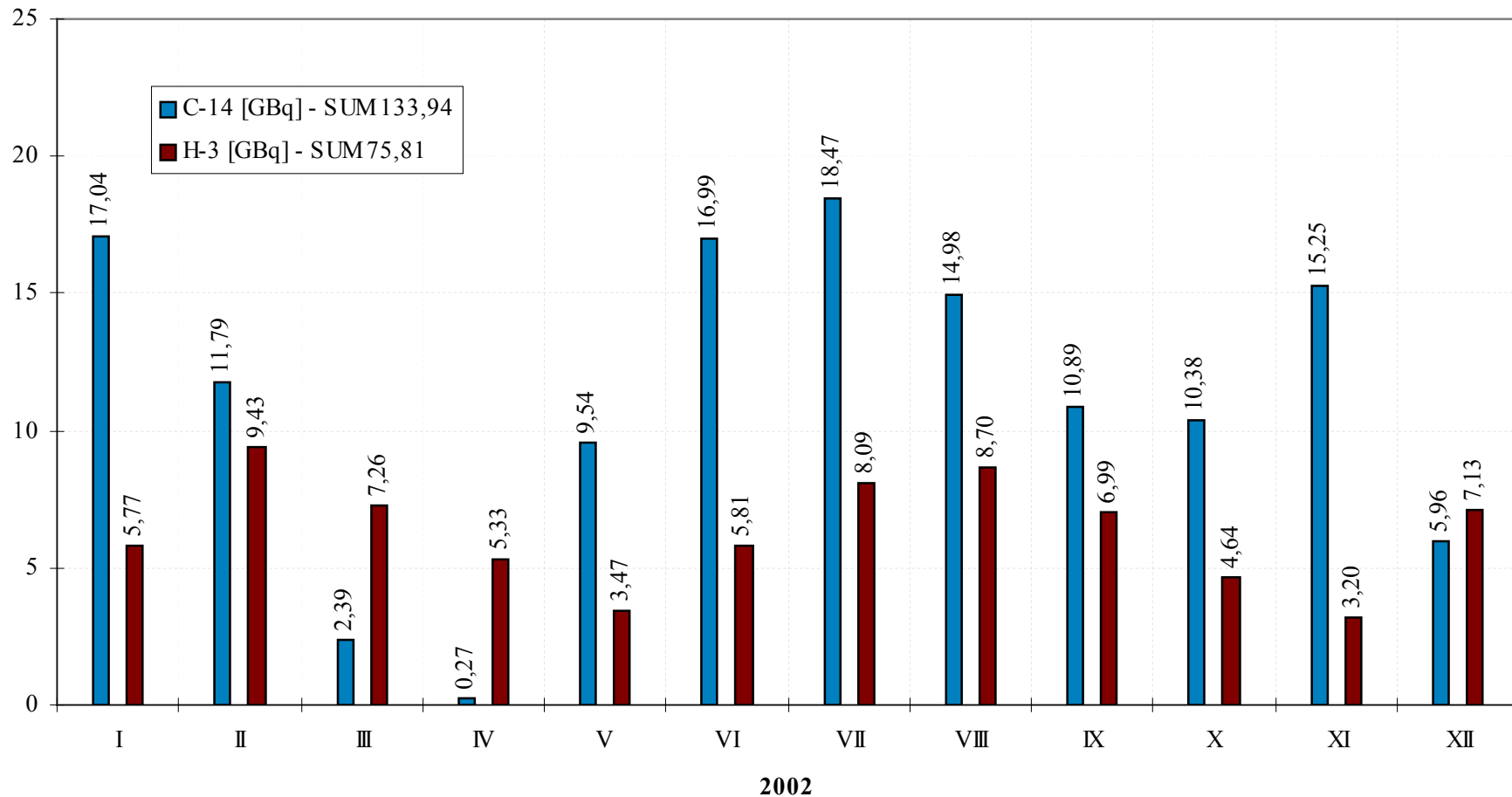
Activity - noble gases, aerosols, iodine





## RESULTS FOR THE YEAR 2002

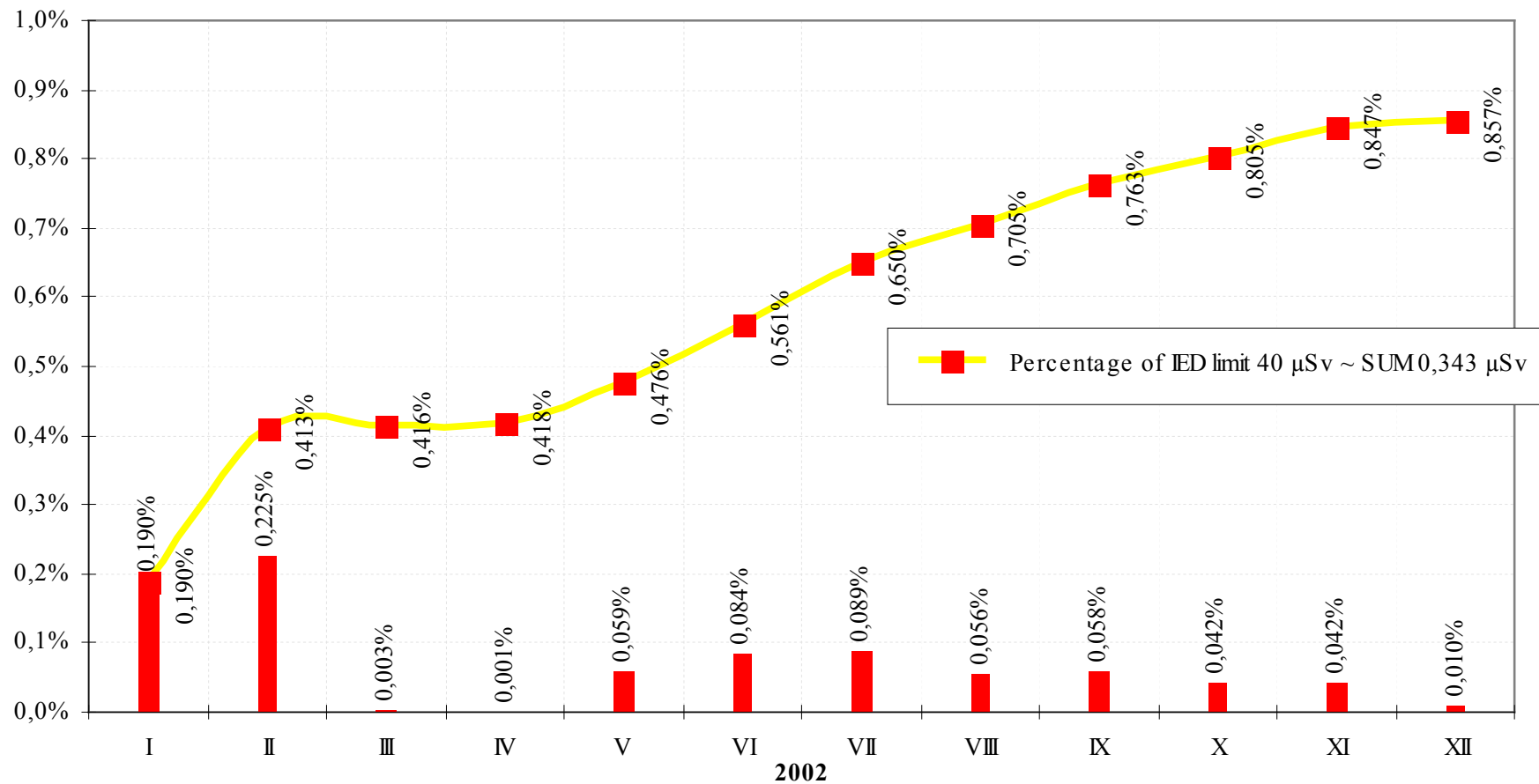
### Activity - C-14, H-3





## RESULTS FOR THE YEAR 2002

### Percentage of IED limit application





### THE MAIN MONITORED QUANTITIES

Auxiliary operations building's control and sampling tanks - prior to every draining

Wastewater tank - every week and month

Volume activity and total activity of released  $\beta$  nuclides (activation and fission products)

Volume activity and total activity of released tritium

Total activity of released  $\beta$  nuclides

Total activity of released  $\alpha$  nuclides



### BALANCING

Through the medium of authorized conversion factors

→ H-3

→ AaFP:  $\gamma$  nuclides, including Sr-89,90 and transuraniums nuclides

→ Balancing calculation - only nuclides with activity > MDA



### H-3

#### Sampling and Liquid scintillation analysis

- Samples from wastewater tank (proportional sampling depending on waste chanel through-flow)
- Samples from control tanks





# ACTIVATION AND FISSION PRODUCTS

## Gamma-spectrometry

- Samples from wastewater tank (proportional sampling depending on waste channel through-flow)
- Samples from control tanks



### RESULTS FOR THE YEAR 2002

#### Limits for released activities (Bq) as for the substances released to the water

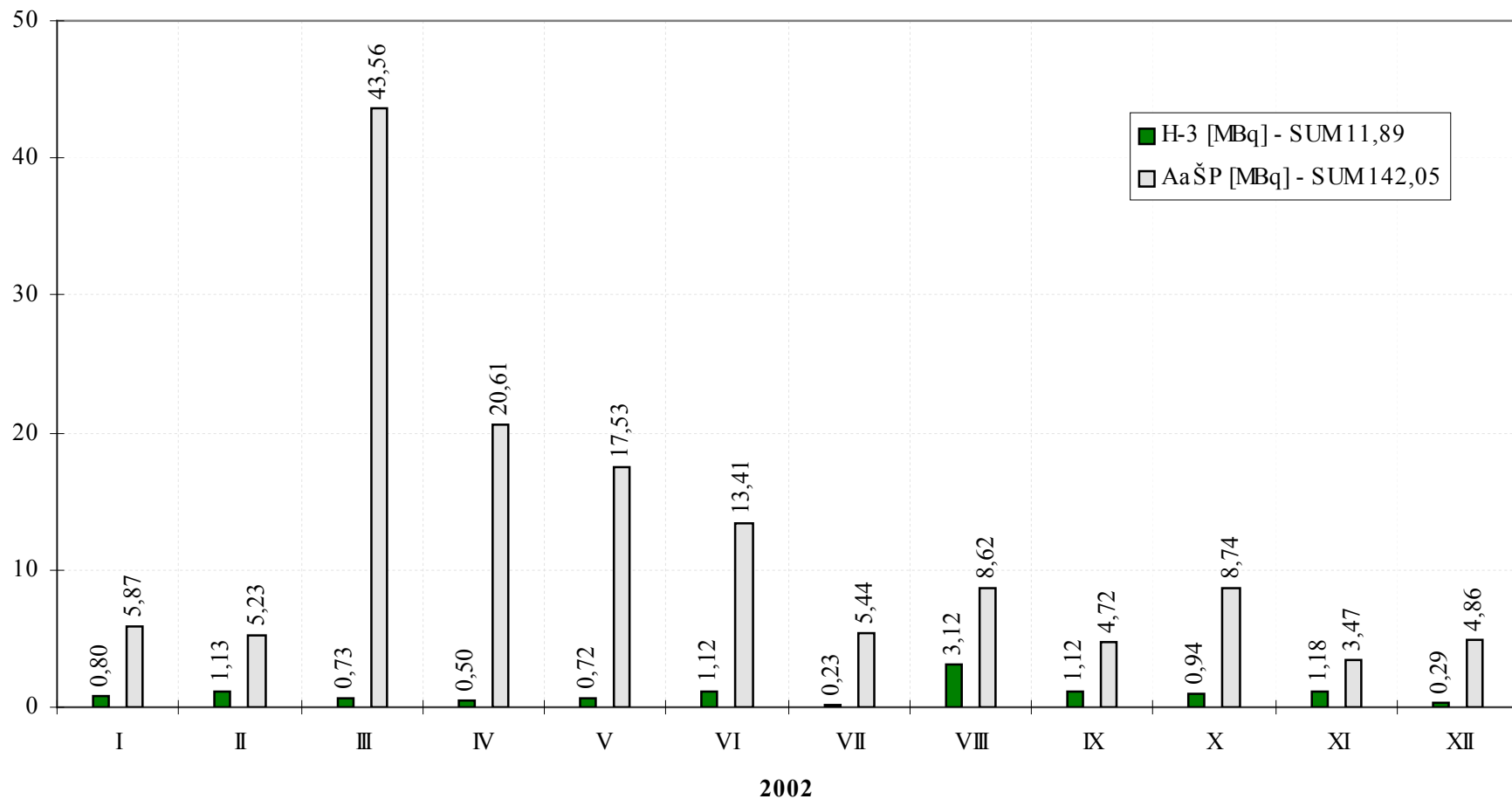
- specified as total released activities:
  - 40 TBq/year for H-3
  - 1000 MBq/year for  $\beta$  nuclides (AaFP)
- specified as radiation bodily harm to an individual from the critical population group ( $\mu$ Sv)
  - 50  $\mu$ Sv/year under the Decree
  - 0,4  $\mu$ Sv/ year under the Technical Specifications for NPP operation

**The limit of 0,4  $\mu$ Sv - exceeded by tens of per cent, expecting up to 80 per cent in the future** (according to the experience gained at the Dukovany NPP and other NPP's in the world)



## RESULTS FOR THE YEAR 2002

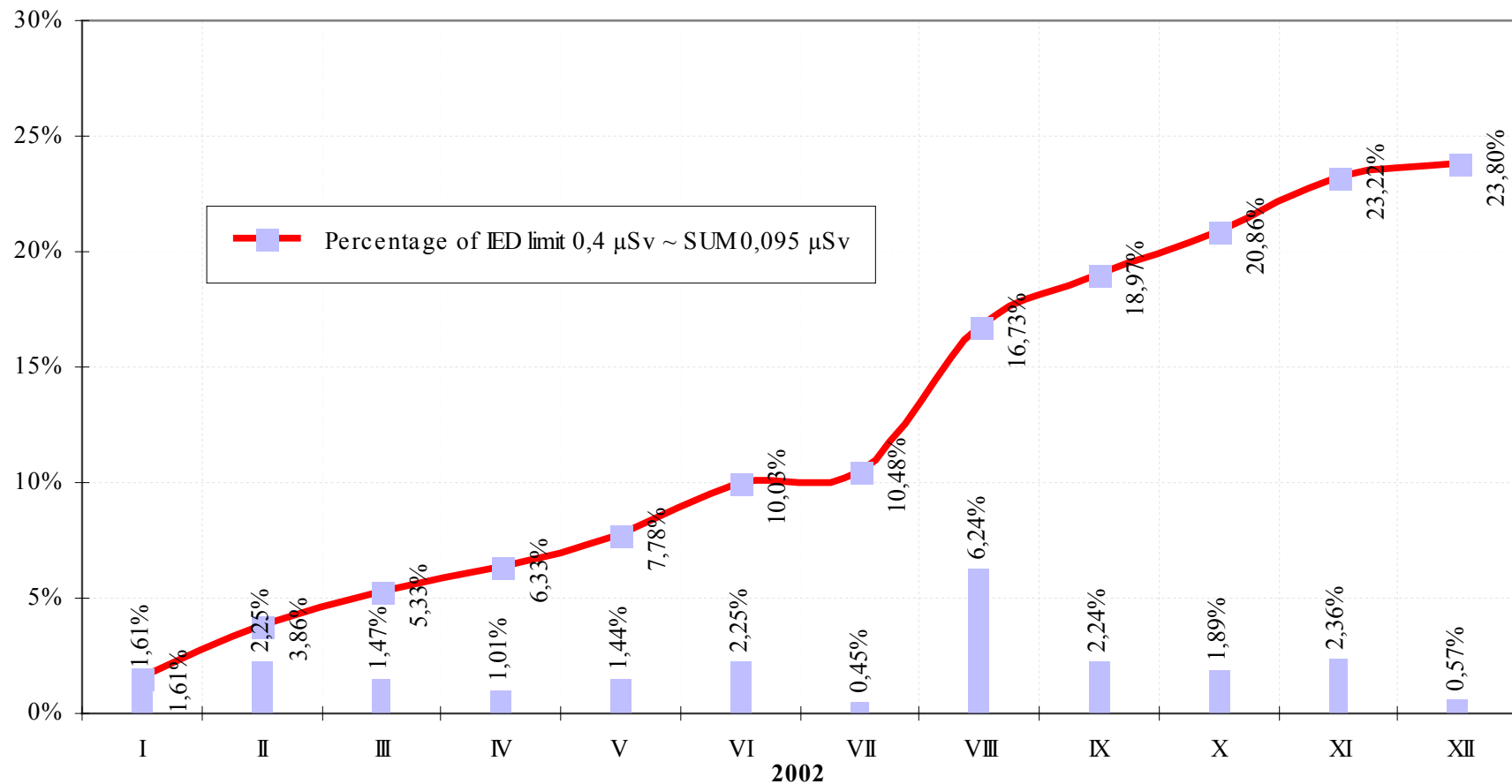
Activity - H-3, activation and fission products





## RESULTS FOR THE YEAR 2002

### Percentage of IED limit application





Releases from NPP Temelín

QUESTIONS AND COMMENTS?

Thank You for your attention, have a nice day!

# **CROATIAN-HUNGARIAN COOPERATION ON THE DANUBE RIVER RADIOACTIVITY MEASUREMENTS**

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## **SUMMARY**

Danube river radioactivity measurements on the border profile Mohač-Batina have been performed since the beginning of 1978 with varying frequency of sampling. Thus, in the period before nuclear power plant Paks started to work joint croatian-hungarian sampling at the border profile was taking place four times a year; the obtained results of measured radioactivity levels were used to assess radioactivity background data. From the start of nuclear power plant Paks running until Chernobyl reactor accident (April 1986) sampling was performed six times a year. After the Chernobyl accident, samples have been taken every month. Since decreased Chernobyl reactor accident influence was estimated until present samples have been taken six times a year.

On the Danube river border profile the concentration activity of gamma radionuclides has been determined in water samples (filtered water and suspended matter), and in fish, sediment and Danube river algae samples.

## **IMPORTANT DATES DURING THE COOPERATION**

**1975** Foundation of the Subcommittee for the Danube river protection against radioactive and thermal pollution (16 meetings held so far).

**1978** First Danube river samples collection

**1979** Foundation of the Experts to the Subcommittee for the Danube river protection against radioactive and thermal pollution (12 meetings held).

**1982** Nuclear power plant Paks started operating (the first block of 440 MWe)

**1987** Nuclear power plant Paks ceased operating, four 1 760 MWe blocks in total

**1991** Yugoslav National Army aggression on Croatia

**1993** Meetings of Croatian and Hungarian watermanagement experts held in Budapest (2<sup>nd</sup> to 4<sup>th</sup> February, 1993) and in Varaždin (20<sup>th</sup> to 23<sup>rd</sup> September, 1993) discussed the issues of watermanagement cooperation between the two countries. The following decision was passed: In accordance with the provisions of the Contract on friendship and cooperation between the Republic of Croatia and the Republic of Hungary, concluded on 16<sup>th</sup> December 1992, until the conclusion of croatian-hungarian agreement on watermanagement issues has been reached, the parties will apply the existing Agreement between the FNR Yugoslavia and NR Hungary concluded on 8<sup>th</sup> August 1955.

**1993** The first Meeting of Croatian-Hungarian subcommission for water quality (11 meetings held so far)

**1993** The first Meeting of the experts to the Croatian-Hungarian subcommission for water quality (10 meetings held so far)

#### **SAMPLING PROGRAM and MEASURING (for one sampling)**

**TABLE 1.** Programs of the measurements radioactivity of the Danube River on border profile

<b>Sample</b>	<b>Type of the measurements</b>	<b>Location of the sampling</b>	<b>Number samples</b>
<b>Water</b>	Total $\beta$		3
<b>Filtered</b>	$\gamma$	Left bank	1
<b>nofiltered</b>	$^{90}\text{Sr}$	The middleRight bank	1
	$^3\text{H}$		1
<b>Fish</b>	total $\beta$		1
<b>Plant-eating</b>	$\gamma$	Danube	1
<b>Meat-eating</b>	$^{90}\text{Sr}$		1
<b>Sediment</b>	total $\beta$	Left bank	2
	$\gamma$	Right bank	2
	$^{90}\text{Sr}$		
<b>Algae</b>	total $\beta$	Object on bank	1
	$\gamma$		

## RESULTS and DISCUSSION

All the measurements performed so far express that the concentrations of long-life fission products activity are significantly reduced compared to the period immediately after the Chernobyl reactor accident. In majority of the results, activity concentration has been reduced to practically the value of “0” level an exception is only the case of  $^{137}\text{Cs}$ , where the activity concentration measured in a sediment sample exceeds the value of “0” level (Fig.1. to Fig.3.).

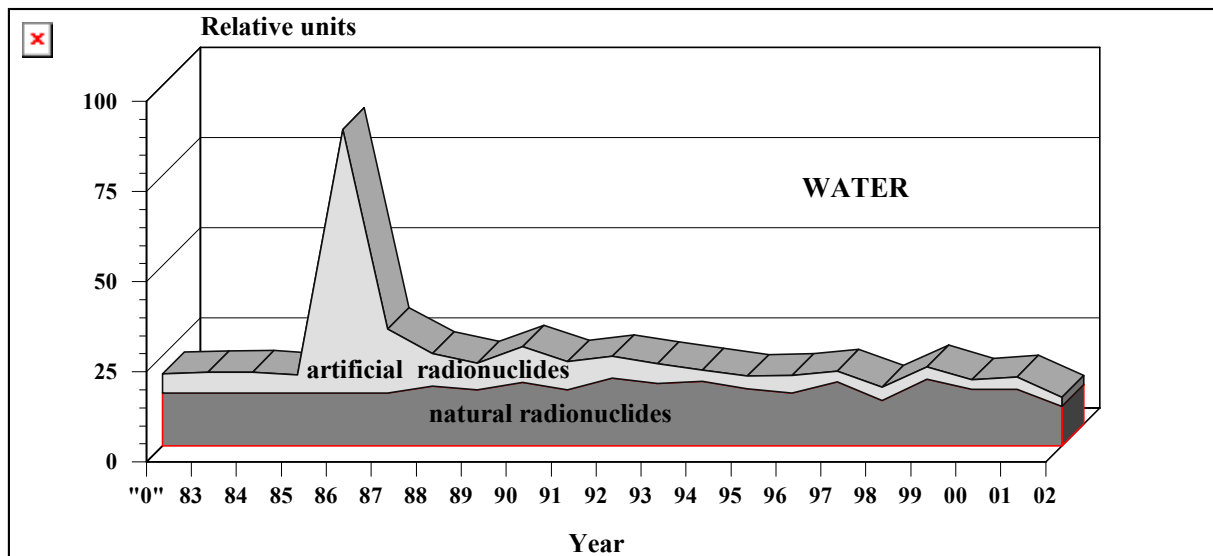


Fig.1. Relationship between artificial and natural radionuclides for water

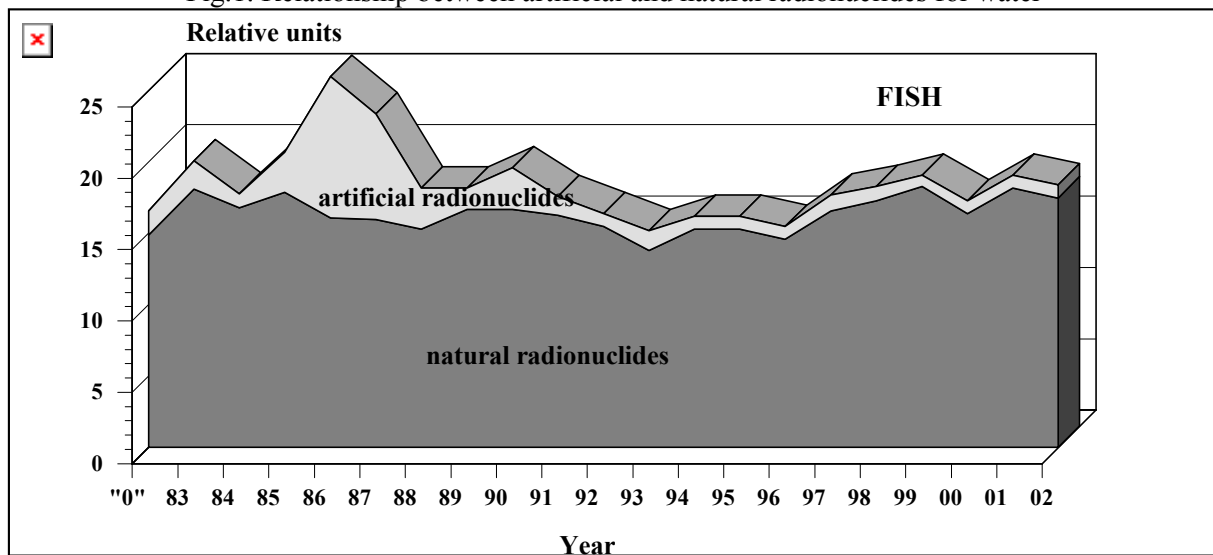


Fig. 2. Relationship between artificial and natural radionuclides for fish



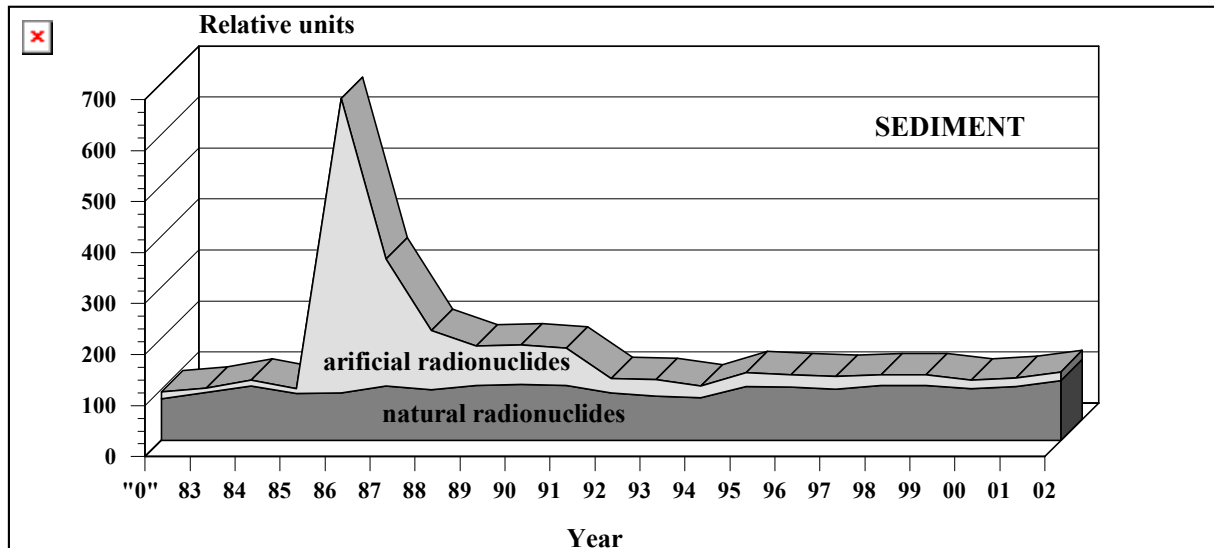


Fig. 3. Relationship between artificial and natural radionuclides for sediment

Irradiation doses for the whole period of measurement of the radioactivity of the river Danube were calculated on the basis of the measurement of the activity of emitted artificial radionuclides and activity of  $^{90}\text{Sr}$  and  $^3\text{H}$  in the samples of the filtered water and measurement of the activity of emitted artificial radionuclides and activity of  $^{90}\text{Sr}$  in the fish tissue. During the calculation of the irradiation doses it was assumed that each individual that live in the vicinity the border profile consumed about 880 L of the Danube water and about 39 kg of the Danube fish. Fish consumption had the biggest influence on the irradiation doses, which could be seen from the presented figure. In the time of Chernobyl accident the irradiation doses were the highest (Fig. 4.). After that time, irradiation doses are decreasing and at present time they are much lower compared to the time of the “zero-point state” determination. It should be noted that calculated doses of the “critical” individual were significantly lower than the allowed dose (1 mSv/year).

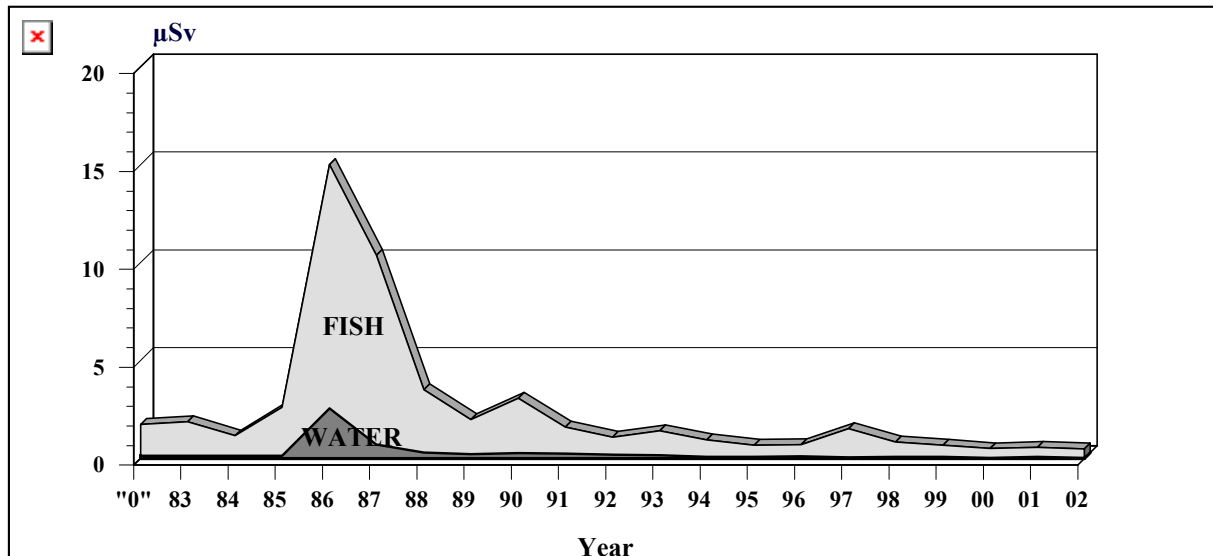


Fig. 4. Annual effective dose, "0" – year 2002 (  $\mu\text{Sv}/\text{year}$ )

## CONCLUSION

Based on those results of activity concentrations measurement, particularly concerning the gamma radionuclides, conclusion can be made that no increase in the radioactivity level in the investigated Danube river samples pertaining to the period of Nuclear power plant Paks work has been observed.

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# **RADIATION MONITORING NETWORK OF THE CZECH REPUBLIC**

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## **Introduction**

The Radiation Monitoring Network (RMN) of the Czech Republic (10 300 000 citizens, 78 900 km<sup>2</sup>) was established after the Chernobyl accident in 1986 and it is developed all the time. It is co-ordinated by the State Office for Nuclear Safety (SONS, [www.sujb.cz](http://www.sujb.cz)) in co-operation with the National Radiation Protection Institute (NRPI, [www.suro.cz](http://www.suro.cz)). At present time a legal basis for RMN is given by the “Atomic Act” No. 18/1997 Coll. and especially by the Regulation of the SONS No. 319/2002 Coll.

RMN performs in two modes of operation: the normal mode, aimed at monitoring of actual radiation situation (including the consequences of previous emergencies such as the Chernobyl accident) and at early detection of radiation accidents, and the emergency mode, aimed at evaluation of consequences of such radiation accident. The normal mode of operation is represented by continuous operation of permanent parts of RMN, in the emergency mode additional parts of RMN are engaged.

Czech RMN consists of the several subnetworks, which include selected or all permanent parts of RMN. The subnetworks are following: the Early Warning Network (EWN), the TLD Territorial Network, the Network of the Measuring Sites of Air Contamination, the Network of Laboratories Equipped with Gamma-spectrometric and Radiochemical Analytical Instrumentation, the Ground and Airborne Mobile Groups. The Laboratory of Monitoring of Internal Contamination and the information system (IS) are a significant part of RMN, too.

## **Early Warning Network of the Czech Republic**

The EWN has been established after 1990 to provide overview information about a radiation situation throughout the territory of CR and to raise an alarm in the case of an

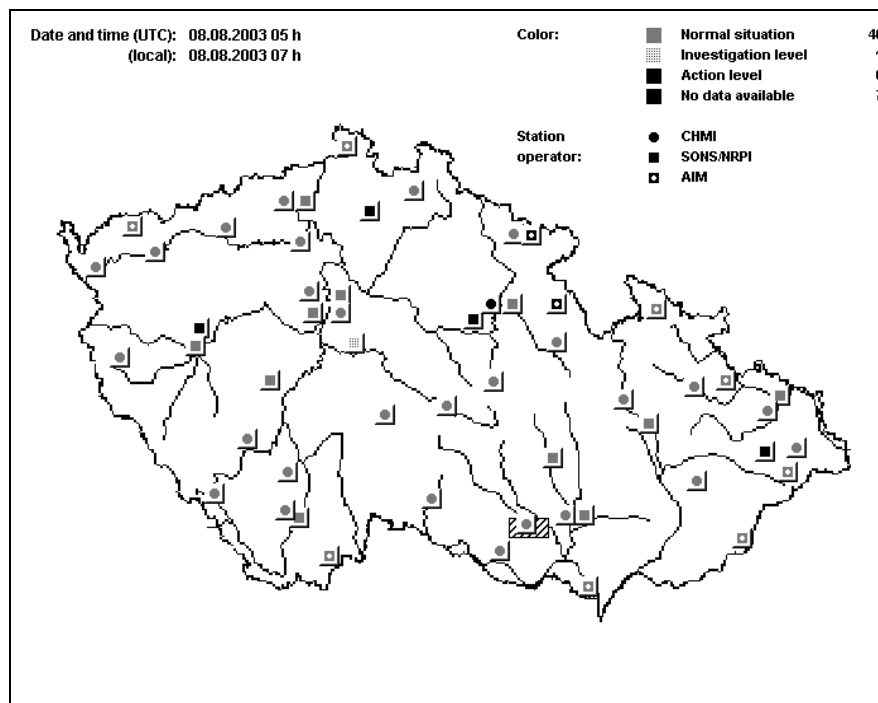
unreported accident outside the territory of CR which would have consequences for the territory of CR.

The EWN measuring sites are equipped with a two-probe system combining of a low-range probe (proportional counter) and a high-range probe (Geiger-Müller counter). The system is able to cover the dose-rate range from natural background up to values in real emergency situation in an energy range from approx. 20 keV to 2 MeV.

The EWN consists of 54 measuring sites covering the whole territory of CR, of which 38 are located at the observatories of the Czech Hydro-Meteorological Institute (CHMI), 9 at the Measuring Points of Air Contamination (MPAC) operated by NRPI and the Regional Centers (RC) of the SONS and 7 at the Measuring Points operated by Fire Brigades (FB).

Each EWN measuring site operates round-the-clock and provides values of average dose-equivalent rate over 10-minutes periods; the measuring sites operated by CHMI provide in addition basic meteorological data (temperature, atmospheric pressure, wind speed and direction, precipitation, dew-point etc.). All data (both radiation and meteorological) are transferred to the Centre of Radiation Monitoring Network (CRMN) located in SONS/NRPI.

Fig. 1: Early warning network – example of the one-hour map



The resulting overview maps (Fig.1), graphs and tables are available to the Crisis Co-ordination Centre (CCC) of SONS. Data are also provided for mutual exchange with Austria and for EURDEP (European Union Radiation Data Exchange Platform) system (via E-mail).

The alarm function of EWN is assured by GSM/SMS-messaging system automatically notifying a person-on-duty through a warning message any time when preset levels of dose-

equivalent rate are reached on any site. A person-on-duty can obtain more detailed information about measured values either using a remote log-in to the server of IS, or via GSM/SMS answering system, check the situation (i.e. whether the increase of dose-rate was caused by fluctuation of natural background) and adopt proper measures.

### **Information System of RMN of Czech Republic**

The IS of the RMN consists of: local applications for collecting of results of monitoring from the monitoring sites, the laboratories etc.; the data transfer system for a transfer of the data from data suppliers to the central servers; the central applications for processing, storing and presenting results of monitoring for the needs of evaluating the radiation situation in CCC, for a mutual exchange of monitoring results on both national and international levels and for publishing an information about a radiation situation.

### **TLD Territorial Network**

A main purpose of the network is to confirm normal radiation situation and to estimate external doses for population in normal and accidental radiation situation. The TLD territorial network consists of 184 monitoring points distributed on the territory of the Czech Republic; it is operated by NRPI and RC SONS. Some of the measuring points also have been installed in adjoining buildings with an intention to obtain information about the shielding effectiveness of the buildings in the case of a radiation accident. In the case of normal radiation situation a three months monitoring period is used. The TLD local networks in the surroundings of two Czech nuclear power plants (NPP) Dukovany and Temelín are operated by NRPI and the local RC, too. As for TL dosimeter, aluminium cups containing plastic cards with three  $\text{Al}_2\text{O}_3:\text{C}$  pellets are used. These cards are read by automatic Dosacus TLD reader.

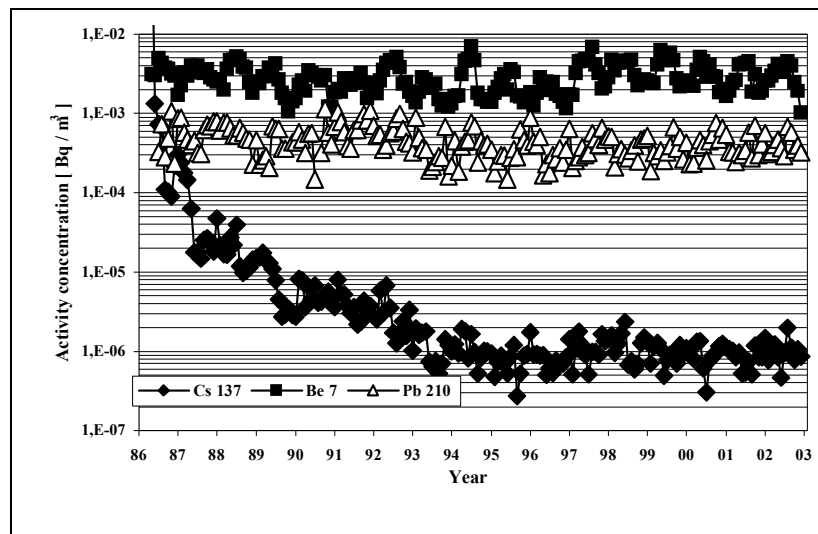
An average quarterly value of photon dose equivalent rate  $\dot{H}_X$  is calculated for each monitoring point, the values of  $\dot{H}_X$  usually range within 70 - 270 nSv/h. The investigation level of 500 nSv/h has never been exceeded and no significant changes for individual monitoring points were observed during recent years.

### **The subnetwork of the measuring sites of the air contamination**

The subnetwork of the measuring sites of air contamination consists of 10 sites equipped, besides other instruments by aerosol samplers with throughput from 40 to 900  $\text{m}^3/\text{h}$ . The samplers work continuously with one-week sampling periods. After a filtration of aerosols a part of the air is led to the cartridge with the sorbent for a sorption of gaseous radioactive iodine. The filters and the sorbent measured without any previous treatment by

semiconductor gamma-spectrometry. The aim of these measurements is not only to detect an abnormal radiation situation in co-operation with EWN, but also to follow long time trends of aerosol activity concentration. Besides artificial radionuclide  $^{137}\text{Cs}$ , natural radionuclides cosmogenic  $^7\text{Be}$  and radon progeny  $^{210}\text{Pb}$  are evaluated by gamma-spectrometry routinely for the purpose to confirm the quality of sampling and measurement in individual laboratories. After gamma spectrometric analysis the radiochemical separation followed by alpha spectrometry or beta measurements is used for the determination of  $^{90}\text{Sr}$ ,  $^{238}\text{Pu}$  and  $^{239,240}\text{Pu}$  in the quarterly combined filters from the measuring site in Prague, the continuously collection of the air for the assessment of activity concentration of  $^{85}\text{Kr}$  in monthly samples is performed here, too.

Fig. 2: The time courses of the mean monthly values of the  $^{137}\text{Cs}$ ,  $^7\text{Be}$  and  $^{210}\text{Pb}$  activity concentration in the air in Prague



The mean activity concentration of  $^{137}\text{Cs}$  in the air in CR was  $1,3 \cdot 10^{-6} \text{ Bq/m}^3$ , of  $^7\text{Be}$   $2,7 \cdot 10^{-3} \text{ Bq/m}^3$  and of  $^{210}\text{Pb}$   $3,4 \cdot 10^{-4} \text{ Bq/m}^3$  and the mean activity concentration of  $^{90}\text{Sr}$  in Prague was  $6,5 \cdot 10^{-8} \text{ Bq/m}^3$  and of  $^{85}\text{Kr}$   $1,4 \text{ Bq/m}^3$  in the year 2002. The time courses of the mean monthly values of the  $^{137}\text{Cs}$ ,  $^7\text{Be}$  and  $^{210}\text{Pb}$  concentration in Prague in the time period June 1986 - December 2002 are presented in Fig. 2.

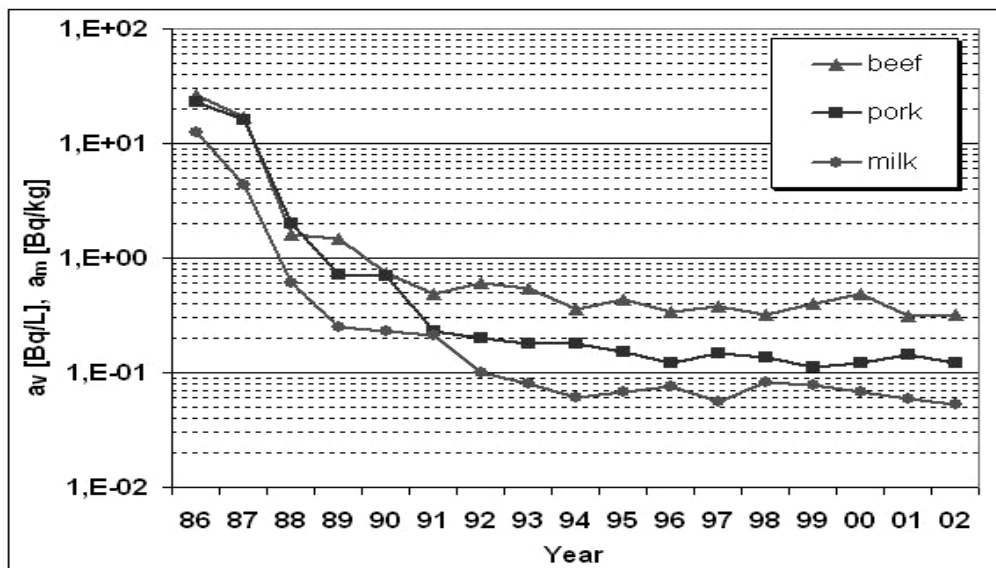
### Network of laboratories monitoring environment samples of foodstuff, water etc.

The network is constituted by 9 laboratories equipped mostly with gamma-spectrometric and radiochemical analytical instrumentation. The frequency of sampling of foodstuff and water is three months or one year (it depends on a sort of foodstuff or on analyzing radionuclide) in the normal mode of operation of RMN. Other environment samples as soils, pastures etc. are collected at request only. Except gamma ray radionuclides the content of  $^{90}\text{Sr}$  is measured in some samples and the content of  $^3\text{H}$  in water, too. To

increase the sensitivity of the determination of very low mass activities in agriculture products, the concentrate methods are used. To increase the representation of the samples, mixed samples from more regions are used.

The average values of the activity of  $^{137}\text{Cs}$  in the year 2002 were in milk 0,053 Bq/L, in beef 0,32 Bq/kg, in pork 0,12 Bq/kg (time courses of yearly average mass, resp. volume activities of  $^{137}\text{Cs}$  are in Fig. 3) and 0,06 Bq/kg in poultry. From kinds of corns from CR the highest value of the activity of  $^{137}\text{Cs}$  was found in oat (0,09 Bq/kg). For some types of samples only range of values of  $^{137}\text{Cs}$  activity was stated; the average wasn't calculated, because the most measured activities were below minimum significant activity (MSA): 0,008–0,7 Bq/kg in vegetable, 0,02–0,05 Bq/kg in fruit; due to the results don't represent the whole territory of CR: 0,6–900 Bq/kg in mushrooms and because of the data number is too low: 0,9–12 Bq/kg in wild berries. In drinking water the activity concentration of  $^{137}\text{Cs}$  was less than MSA  $3 \cdot 10^{-4}$  Bq/L (on 95% confidence level), the activity concentration  $^{90}\text{Sr}$  was  $4 \cdot 10^{-3}$  Bq/L and of  $^3\text{H}$  1,3 Bq/L. The average value of  $^{90}\text{Sr}$  activity in milk was 0,06 Bq/L and in corn 0,15 Bq/kg. All activities are in fresh weight/volume.

Fig. 3 Average year mass/volume activity of  $^{137}\text{Cs}$  in pork, beef and milk



### Ground and Airborne Mobile Groups

The mobile groups are carried on at each RC SONS, NRPI, departments of Ministry of Interior and of Ministry of Finance, NPPs Temelín and Dukovany. The mobile groups provide for mainly: mapping of radiation situation on the base of ground or airborne exploration; distribution of TL dosimeters; qualitative and quantitative assessment of radionuclides in a field (in-situ spectrometry); sampling of aerosols in a field; sampling of environment samples

(soil, vegetation, water etc.) and search of orphan's sources of ionizing radiation in the environment.

The mobile groups regularly take part in exercises in the Czech Republic, the mobile group of NRPI in international exercises, too.

### **Monitoring of the internal exposure and whole body counting**

The national survey of the internal exposure by  $^{137}\text{Cs}$  is carried out by means of measurement of the activity of  $^{137}\text{Cs}$  excreted in 24 hours urine. Samples are collected every year, usually during May and June from about 70 persons whose diet represented roughly with their diet general population of CR. The average activity of  $^{137}\text{Cs}$  excreted in 24 hours urine was 0,48 Bq in the year 2002. According to that, the calculated average content (retention) of the  $^{137}\text{Cs}$  activity in a human body was approx. 79 Bq. The estimation of the committed effective dose based on this national survey was 2,9  $\mu\text{Sv}$   $^{137}\text{Cs}$ .

A monitoring of internal exposure by  $^{137}\text{Cs}$  in a reference group of about 30 persons, mostly Prague citizens, is performed on the whole body counter (WBC) in the NRPI in Prague every year. The average activity of  $^{137}\text{Cs}$  in a person was estimated to be about 80 Bq in 2002.

### **Conclusion**

The data of RMN resulting from monitoring are transferred to the central database of CRMN, processed by the IS. They are used in normal and emergency situations for an evaluation of radiation situation and for preparation of recommendations for protection of the public and the environment. Selected results of monitoring are continuously published on web site of NRPI and summarised in the Annual Reports on Radiological Situation on the territory of the Czech Republic, issued by the NRPI.

In 2002 any extraordinary radioactivity in the environment was not detected and also none of the measuring points recorded any exceeding of established investigation levels. In components of environment and also in human beings a very low activity of  $^{137}\text{Cs}$  was still measurable, that had been released into environment after the Chernobyl accident and by the nuclear weapon tests in sixties of the last century.

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# TELEDOSIMETRY SYSTEM OF MOCHOVCE NPP

Stefan Grubel, Mochovce NPP

The task of teledosimetry system (TDS) is the continuous measurement of gamma dose rates as well as activity concentrations of aerosols and iodine.

TDS is spaced on 2 circles

- ❖ on-site, around the buildings and
- ❖ off-site, covering the radiological most important locations in the planning zone of emergency preparedness

**On-site** monitoring posts, inner circuit there are installed 16 detectors for measuring dose rates in a circle line inside the power plant site at the distances between approximately 200 m and 520 m from the exhaust air stack of Mochovce 1,2. The technical versions of all these measuring posts are from the same type

On-site are installed 3 large containers for measuring gamma dose rate and activity concentration of aerosols and iodine too.

**Off-site** monitoring posts at places of living

consist of 16 small containers with dose rate measurements and iodine sampling units and 8 large containers with dose rate measurement and aerosol and iodine monitoring unit. Three of these large containers are installed nearby the NPP inside the fence of the NPP area. The technical version of these measuring posts are from the same type. Five of these containers are installed faraway of the NPP outside the fence of the NPP. These five posts are from the same type.

## Data transmission

The on-site TDS monitoring posts are connected by means of cable. The cabling consist of two cable circuits and are performed with shielded cables. The enlarged monitoring posts (large containers) which are located on-site will also be connected via cables.

The data transmission from the off-site TDS detectors and enlarged monitoring posts are performed in two different ways. The data transmission are first performed via radio transmission. In case of problems with or failure of the radio transmission the monitoring posts is possible switch to mobile phone connection.

The process data is continuously acquired, stored and processed by the Central Radiological Computer System of the power plant. The TDS data are a main part of the radiological information system, which continuously provide the information of the measurements and evaluate possible radiological consequences.

# SOME CRITICAL POINT ON $^{90}\text{Sr}$ DETERMINATION

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## INTRODUCTION

Determination of radioactive strontium is a part of the control of quality of air, water, soil and food that is performed within the nuclear facility monitoring. Its determination is time consumed and complicated because of radio-chemical properties. Determination requires the chemical isolation of strontium from the sample and detection on proportional gas counter or liquid scintillation counter. Therefore, the determination methodology has two stages; the first stage is chemical isolation, and the second stage is detection and quantification.

For radiostrontium determination numerous methods has been developed. They usually have different isolation procedures, while the detection is almost the same. However, the exactness and reliability of the obtained results are often questionable, especially when the concentrations are very low (but this sometimes applies to higher concentrations too), because the obtained results can vary to a large degree, so it is hard to determine the correct result. This happens because of the complicated determination procedure and different methods used. To resolve this issue, the methodologies are often harmonized, as far as possible, by means of inter-comparative measurements.

This paper will examine the results of inter-comparative measurements of four laboratories that participate in the program of monitoring the nuclear plant at Krško, with a special presentation of all factors that impact the exactness and reliability of final results. Inter-comparative measurements are done each year and results of the last three years (1) will be presented. Radioactive strontium ( $^{90}\text{Sr}$ ) was measured in the samples of sediment, soil, milk, fish and ashes – a total of 11 samples.

## RESULTS AND DISCUSSION

Since the report on the measurement results does not include the description of methods used, it is impossible to discuss specific methods. It is only possible to comment the results according to the types of samples and level of radioactivity in a specific type of sample. Of course, this can be viewed from the critical standpoint of years of experience in measuring  $^{90}\text{Sr}$  in natural samples and of knowing all the issues related to

Table 1. Results of Inter-comparative measurement

Sample/location	Laboratory 1	Laboratory 2.	Laboratory 3.	Laboratory 4.
Activity <sup>90</sup> Sr (Bq/kg)				
Soil "Amerika"	2.80 0.30	3.50 0.70	2.30 0.40	3.10 0.20
Soil "Kosova vrbina"	1.50 0.30	1.40 0.40	1.20 0.20	1.60 0.20
Ash "Videm Krško"	460 30	370 40	350 4	---
Sediment "Baja"	0.31 0.09	---	0.75 0.10	0.52 0.11
Sediment " Mohacs"	0.25 0.10	---	1.60 0.30	0.24 0.07
Fish1 "Mohacs"	1.40 0.20	---	1.40 0.30	---
Fish1 "Mohacs"	1.90 0.20	---	1.80 0.30	---
Soil "Kobarid"	15.0 1.00	13.0 1.00	17.0 3.00	12.00 1.00
Milk "M. Sobota"	1.20 0.10	1.20 0.20	1.20 0.20	1.00 0.10
Sediment "Sora"	0.50 0.10	0.60 0.05	0.30 0.20	0.22 0.02
Sediment Todrašćica"	0.70 0.10	0.80 0.09	0.60 0.30	0.40 0.04

different methods, depending on the type of sample. As already mentioned the determination methodology has two stages, and it will be commented all the issues in that light. The formula for calculating activities will be used to better understand all the factors impacting the result (4). The formula for calculating activities is:

$$A_{90\text{Sr}} = (C_1/T_1 - B_2/T_2) / (V * E * I_{\text{Sr,Y}})$$

Where:  $A_{90\text{Sr}}$  is activity concentration of <sup>90</sup>Sr (Bq/kg, m<sup>3</sup>),  $C_1$  is measured counts of the sample,  $T_1$  is the sample-counting time (s),  $B_2$  is the background counts,  $T_2$  is the background counting time (s),  $V$  is sample quantity (kg, m<sup>3</sup>),  $I_{\text{Sr,Y}}$  is recovery of strontium or yttrium

It may be concluded that the reliability of results is directly dependent on the number of counts that themselves depend on recovery and detector efficiency. It is a known fact that

the results will be more reliable if a larger number of counts are recorded. As mentioned before the number of counts depends on chemical recovery and detector efficiency, and it is proportionate to these values. The given instrument usually has a constant efficiency. It should be mentioned that the background level also impacts the exactness, and the detection limit depends on it. If background level is lower the detection limit is lower and the determination is easier and more reliable. Therefore reliability of determination depends on the instrument and environment.

In the first stage of the determination methodology chemical isolation of strontium from a natural sample is important because recovery (and number of counts) of strontium depends on this stage. The success of the isolation of strontium depends on the type of sample. When the samples are easy to process chemically, i.e. (water, fish, milk), the process is quicker and the recovery usually is higher. The isolation of strontium from samples that are hard to process, such as the samples of sediments and soil, is linked with many difficulties. Various acids need to be used to extract as much strontium as possible in order to have the highest possible recovery, which is extremely difficult. As already mentioned recovery impacts on other factors in the subsequent determination process. The next step is the quantity of the sample being processed. It depends on the level of activities and the features of the detector, such as the background and the detector efficiency. The higher activity of the sample, and the lower background and the higher detector efficiency, the lesser sample quantity can be used. Of course, measurement time must also be adapted to the determination requirement (lower limit of determination). Let us say that we have a detector with the background of 0.9 count/min, with the 33% efficiency, and we assume that we want to determine the activity of 1 Bq with a statistically relevant reliability (5). Then, with the 60% recovery of strontium and a measurement time of 100 min., we must process 20 g of sediment or soil, 55 l of water or 450 g of fresh fish.

Strontium determination can be performed by measuring  $^{90}\text{Sr}$ - $^{90}\text{Y}$  in radiochemical equilibrium or by measuring  $^{90}\text{Y}$  after separating it from  $^{90}\text{Sr}$ . Each of these methods has its strengths and weaknesses (6). If we measure  $^{90}\text{Sr}$ - $^{90}\text{Y}$  in equilibrium, the sample must be left for 14 days. The advantage of this method is a more elegant measurement, because the sample does not need to be measured immediately after the chemical separation. If  $^{90}\text{Y}$  is separated from sample, it must be measured very soon after the separation, because  $^{90}\text{Y}$  has a short half-life (64 hours), so the activity drops quickly. Still, it has the advantage of a higher measurement efficiency compared to the measurement in equilibrium, because  $^{90}\text{Y}$  has the energy of 2.2 MeV. The presence of  $^{89}\text{Sr}$  in sample complicates the determination procedure

even more, because it requires the separation of yttrium from strontium after attaining of equilibrium (determination of  $^{90}\text{Sr}$  through  $^{90}\text{Y}$ ).

We have indicated the important problems arising when  $^{90}\text{Sr}$  is being determined in natural samples. The results in Table 1 show good consistency of results for specific types of samples, but also the deviations of the sediment samples. As we already said, the results may largely vary for both higher and lower activities. This happens more often in case of low concentrations of  $^{90}\text{Sr}$  in the sediment samples, which are not easy to dissolve. Numerous factors that create difficulties in this segment result in low isolation recovery and low reliability of the final result and the possible great variation of results. This can be observed in higher activities as well. The opposite applies to the results of determining  $^{90}\text{Sr}$  in samples that are much easier to process (milk, fish), where all four laboratories showed good consistency even for lower concentrations.

## CONCLUSION

We can conclude that the determination of radioactive strontium in natural samples is a very demanding task that requires the engagement of well-trained professionals. In order to obtain results that are as reliable as possible, it is necessary to continuously and actively follow all the improvements in the determination methods and to verify the work by participating in inter-comparative measurements.

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# THE STUDY OF AEROSOL COMPONENT OF ATMOSPHERE IN BRATISLAVA

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## Introduction

Atmospheric aerosols are submicron particles, on which the elements of solid or liquid state present in the atmosphere are captured. Also radioactive nuclides are being bound to them. Aerosol particles are sorted according their size into small-dispersive ( $d < 0.1 \mu\text{m}$ ), middle-dispersive ( $0.1 < d < 10 \mu\text{m}$ ) and big-dispersive ( $d > 10 \mu\text{m}$ ). The latter are staying only short time in the atmosphere, due to gravitation they are quickly falling on the Earth's surface.

Most important radionuclides present in the low-level atmosphere are  $^{222}\text{Rn}$ , its short-lived and long-lived progeny  $^{210}\text{Pb}$  and  $^{210}\text{Po}$ .  $^{222}\text{Rn}$  originates from the decay chain of  $^{238}\text{U}$ , which is present in terrestrial crust. In comparable concentration to  $^{210}\text{Pb}$  and  $^{210}\text{Po}$ , the cosmogenic  $^7\text{Be}$  is present in the low-level atmosphere. Antropogenic  $^{137}\text{Cs}$ , coming mainly from nuclear bomb tests and Chernobyl accident, and primordial  $^{40}\text{K}$  are present in very low concentrations in the atmosphere.

The aim of this study is to continue in the long-term measurement of radioactivity of aerosol component in low level atmosphere in Bratislava [1]. In particular our work is concerned with  $^{210}\text{Pb}$ ,  $^7\text{Be}$ ,  $^{137}\text{Cs}$  and  $^{40}\text{K}$  measurements.

## Experimental

Aerosol particles in the atmosphere have been collected using the nitro-cellulose filters (PRAGOPOR – with hole diameter  $0.8 \mu\text{m}$ ) with the collection efficiency of approximately 100 %. Aerosol samples have been collected at Meteorological Station near the Faculty of Mathematics, Physics and Informatics in Bratislava. After exposing in a sampler device the filters have been measured by a semiconductor HPGe detector with a beryllium window placed in the low-background shield. Corrections for radioactive decay were applied on the obtained values.

In case of  $^{137}\text{Cs}$  and  $^{40}\text{K}$  because of their very low concentrations two or more sets of samples were accumulated together to obtain flow volumes on  $10000\text{ m}^3$  level. These accumulated samples have been measured by a large volume HPGe detector with higher detection efficiency, than in the first case, placed also in low background shielding.

Simultaneously, the local meteorological data (wind velocity, air temperature, air pressure, air humidity and amount of precipitation) have been recorded.

## Results and Discussion

During monitoring period from March 2001 to April 2003 68 sets of nitro-cellulose filters were sampled. Measured concentrations of  $^{210}\text{Pb}$  and  $^7\text{Be}$  over the term are presented in figure 1. The concentrations of  $^{210}\text{Pb}$  ranged from  $0.27$  to  $2.93\text{ mBq}\cdot\text{m}^{-3}$  with a mean value  $0.87 \pm 0.02\text{ mBq}\cdot\text{m}^{-3}$ . The concentrations of  $^7\text{Be}$  ranged from  $0.46$  to  $4.42\text{ mBq}\cdot\text{m}^{-3}$  with a mean value  $2.14 \pm 0.04\text{ mBq}\cdot\text{m}^{-3}$ . Both radionuclides show seasonal variations.  $^{210}\text{Pb}$  reach higher values in autumn and winter months, what is attributed to a frequent inversion conditions of the surface air layers [7]. The decrease of the  $^{210}\text{Pb}$  concentration in warm

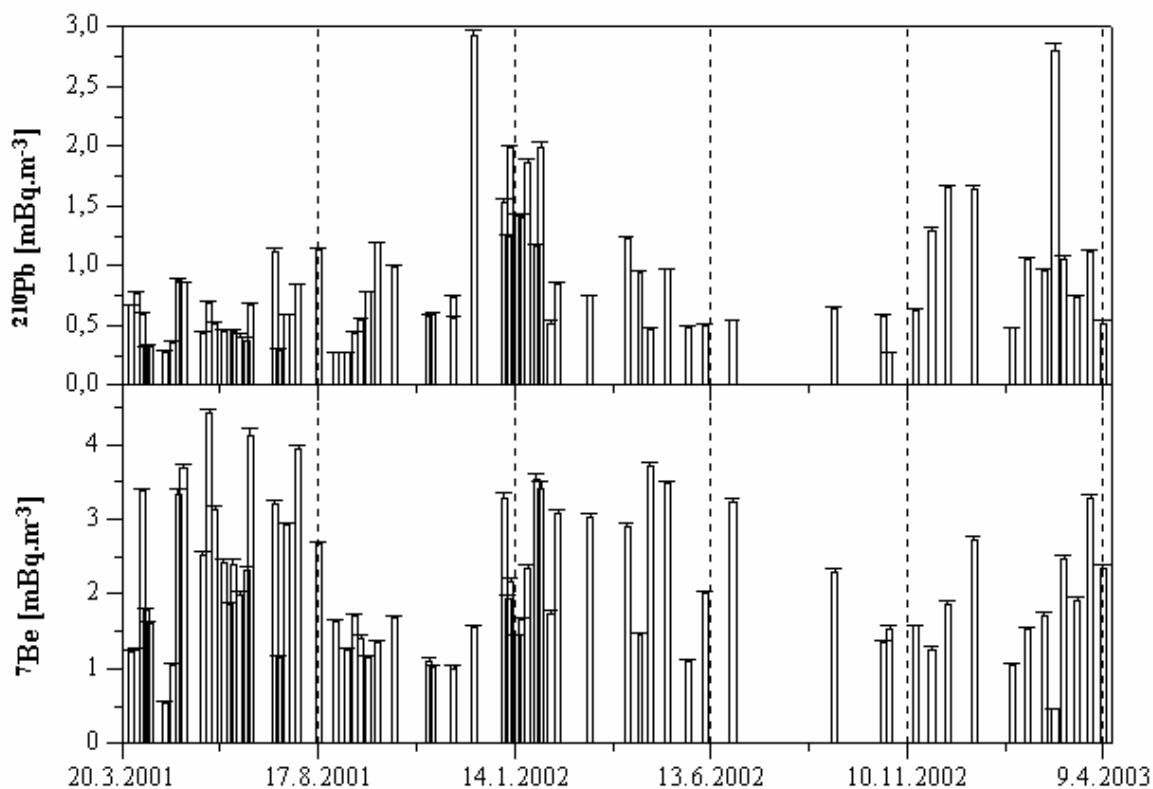


Fig. 1. Air activity concentrations of  $^{210}\text{Pb}$  and  $^7\text{Be}$  in aerosol component of the atmosphere.

spring and summer season is a result of intensive air mixing. The concentration of  $^7\text{Be}$  behaves almost inversely to the  $^{210}\text{Pb}$  concentration. The highest values of activity have been

detected during spring and summer term. It is an effect of air-mass transport from stratosphere to troposphere. On the contrary in cold months these exchange processes are reduced and as a result the supply of  $^7\text{Be}$  produced in higher layers of atmosphere declines.

Correlation coefficients between air concentrations of radionuclides and meteorological parameters are presented in table 1. Relatively strong correlation between the concentrations of  $^{210}\text{Pb}$  and air pressure is observable. This result is astonishing considering between radon progeny and the atmospheric pressure the opposite trend was observed [6]. Also relatively strong anti-correlation of air temperature with the  $^{210}\text{Pb}$  concentration is apparent. The increasing air temperature induced stronger air mixing [8] what effects remarkable decrease of  $^{210}\text{Pb}$  concentration. In case of the  $^7\text{Be}$  there was not found out any significant correlation with meteorological parameters.

Table 1

Correlation coefficients between air concentrations of radionuclides and meteorological parameters

	Wind velocity	Air temperature	Air pressure	Air humidity	Precipitation
$^{210}\text{Pb}$	-0.30	-0.51	0.59	0.35	-0.22
$^7\text{Be}$	0.07	0.25	0.07	-0.39	-0.21

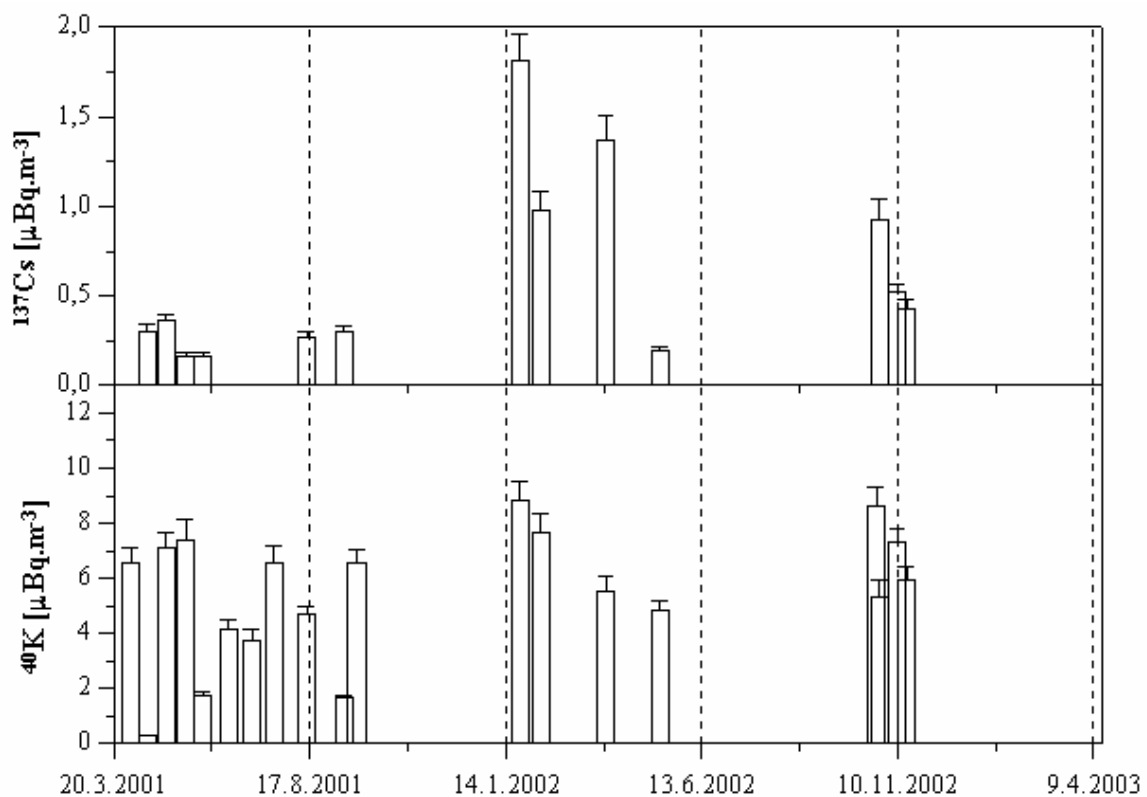


Fig. 2. Air activity concentrations of  $^{137}\text{Cs}$  and  $^{40}\text{K}$  in aerosol component of the atmosphere.



$^{137}\text{Cs}$  and  $^{40}\text{K}$  in atmospheric aerosol have very low concentrations. The results of air activity concentration measurements are presented in figure 2. For  $^{137}\text{Cs}$  only 13 and for  $^{40}\text{K}$  only 19 samples reached the activity concentration above detection limit. Values of the activity concentration of  $^{137}\text{Cs}$  ranged from 0.16 to 1.81  $\mu\text{Bq}\cdot\text{m}^{-3}$  with mean value 0.6  $\pm$  0.06  $\mu\text{Bq}\cdot\text{m}^{-3}$ . For  $^{40}\text{K}$  the concentration ranged from 0.28 to 8.83  $\mu\text{Bq}\cdot\text{m}^{-3}$  with mean value 5.5  $\pm$  0.46  $\mu\text{Bq}\cdot\text{m}^{-3}$ .

Also the mass of dust content for every filter sample have been measured. The results are shown in figure 3 and the values ranged from 0.23 to 134  $\mu\text{g}\cdot\text{m}^{-3}$ , what are typical amounts for European area [2].

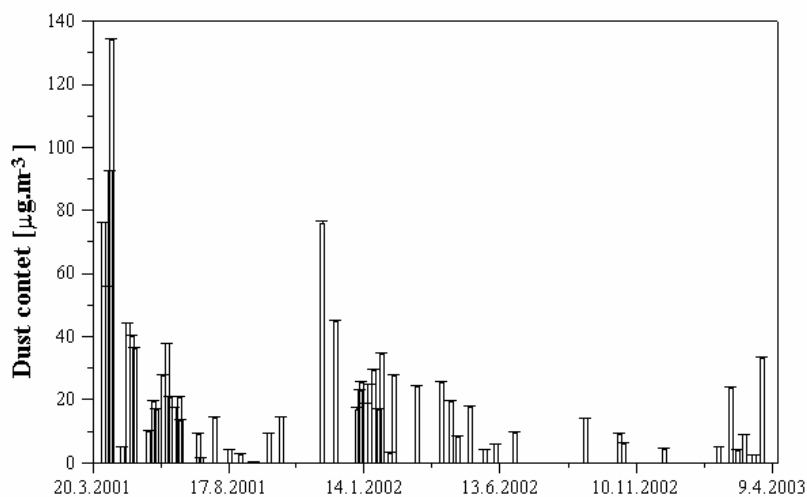


Fig. 3. Dust content collected on aerosol filters.

## Conclusion

The concentrations of  $^{210}\text{Pb}$  and  $^7\text{Be}$  ranged from 0.27 to 2.93  $\text{mBq}\cdot\text{m}^{-3}$  with a mean value 0.87  $\pm$  0.02  $\text{mBq}\cdot\text{m}^{-3}$  and from 0.46 to 4.42  $\text{mBq}\cdot\text{m}^{-3}$  with a mean value 2.14  $\pm$  0.04  $\text{mBq}\cdot\text{m}^{-3}$ , respectively. Both radionuclides show seasonal variations. The concentrations of  $^{137}\text{Cs}$  and  $^{40}\text{K}$  are near detection limit. It is a reason why only few data have been evaluated. The mean value of  $^{137}\text{Cs}$  and  $^{40}\text{K}$  is 0.6  $\pm$  0.06  $\mu\text{Bq}\cdot\text{m}^{-3}$  and 5.5  $\pm$  0.46  $\mu\text{Bq}\cdot\text{m}^{-3}$ , respectively. Activity of aerosol component of low level atmosphere in Bratislava shows typical values of air activity concentrations of  $^{210}\text{Pb}$ ,  $^7\text{Be}$ ,  $^{137}\text{Cs}$  and  $^{40}\text{K}$  for European area [2, 3, 4 and 5].

## Acknowledgements

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# RADIOACTIVITY MONITORING OF THE DANUBE IN AUSTRIA

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**Abstract** - The atmospheric nuclear weapon tests and the nuclear power production activities have led to an increasing radioactive contamination of the environment on a global scale since the middle of the 20<sup>th</sup> century. In this paper the long-term development, radioecological behaviour and environmental impact of long-lived anthropogenic and natural occurring radionuclides in the Danube's water / sediment ecosystem within the last fifteen years are presented by way of <sup>137</sup>Cs. The results are discussed in regard to risk assessment, radioecological modelling and environmental research.

## **Introduction**

The Austrian section of the Danube has been intensively monitored since 1986 in order to investigate the behaviour of natural as well as anthropogenic derived radionuclides in the river compartments. The monitoring started in the month after the Chernobyl accident, as parts of Austria have been heavily contaminated by radioactive fallout. The monitoring, though, did not only emphasise on fallout radionuclides like <sup>137</sup>Cs but also took into account the presence of naturally derived radionuclides such as <sup>7</sup>Be and <sup>210</sup>Pb.

The aim of the project is to learn more about ecological long-term behaviour of radionuclides in rivers and catchments and their relations to climate and soil.

## **Methods**

In order to obtain a comprehensive data set, it was necessary to carry out a continuous and temporal well defined monitoring scheme along the Danube.

Hence, sediment samples have been taken by a sediment traps in the cooling circuits of the power stations at the locations Ottensheim/Wilhering (river km 2146.7),

Wallsee/Mitterkirchen (r.km 2094.5) and Greifenstein (r.km 1949.2) and additionally at Nussdorf/ Vienna (river km 1933.3) since 1989 on a monthly basis.

Moreover, water samples, including sediment load, have also been taken on a monthly basis from 1989 to 1995 in Ottensheim, Wallsee and Nussdorf and since 1990 in Greifenstein. Flood and high water periods have been documented by additional samples taken each 6 to 24 hours during a high water event.

Suspended particles have been separated from water by the use of a centrifuge. Enrichment of radionuclides from particle-free water samples, necessary for radiometric analysis, was carried out by the use of a high-volume vacuum-rotation-retort.

The sediment samples have been air-dried, homogenised and subsequently split up for the different designed analysis (radiometric, granulometric and chemical),

The refined water, particle and sediment samples have thereafter been radiometrically analysed using Germanium detectors for low-level gamma-spectrometry. Moreover, a grain size distribution of the sediments has been created. Water contents and organic matter have been determined by drying at 105°C and 360°C, respectively.

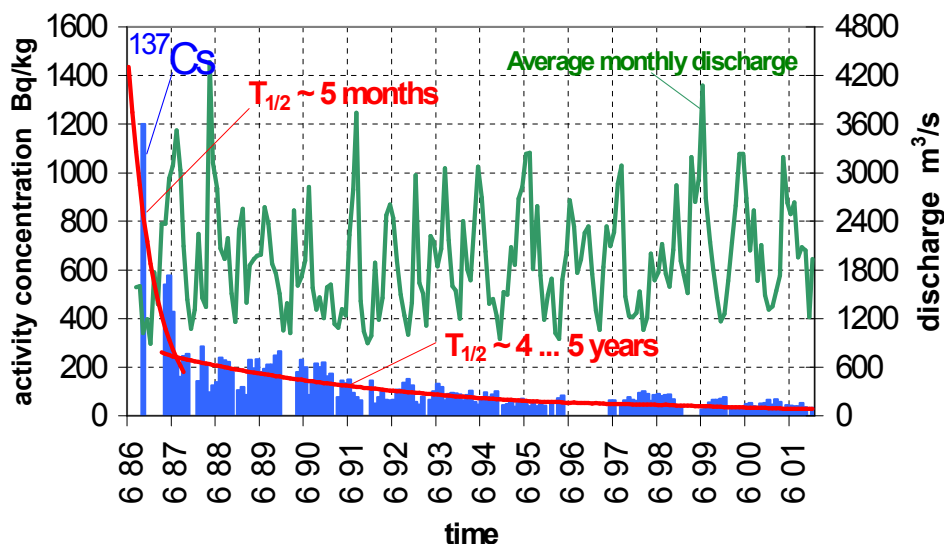
Influence of grain size distribution, organic contents and geo-chemical composition (Maringer 1996, 1996b) of the sediment samples have been levelled out by applying empirically derived factors on the original analysis results. This was necessary in order to be able to compare samples from different location, taken at different points of time with each other.

## **Results**

### ***<sup>137</sup>Cs in sediment samples***

Sediment samples, taken on a monthly basis from the four sampling locations, show very similar trends. Peak <sup>137</sup>Cs activity concentrations have been reached in the months following the Chernobyl disaster. The subsequent decrease of activity concentrations can be approximated by an exponential function for the first two years. This rather rapid decay is characterised by an ecological half-life (describing the time the radionuclide remains in the hydrological system) of around 5 months. From 1988 on, the decline of <sup>137</sup>Cs activity concentration describes a much more flat, though still exponential graph, suggesting an ecological half-life of around 5 years (Fig.1). The sudden change in the decrease of radionuclides cannot not be explained totally at the present. But there are strong indications of increasing <sup>137</sup>Cs adsorption processes in the catchment soil particles in the first weeks/ months

after the environmental contamination. Some authors investigated slow diffusion of  $^{137}\text{Cs}$  into deeper and stronger boundary positions of illitic clay mineral lattice (Smith 1999).



**Figure 1:**  $^{137}\text{Cs}$  activity concentration in sediment between 1986 and 2001, Danube, Nussdorf/ Vienna (river km 1933,2)

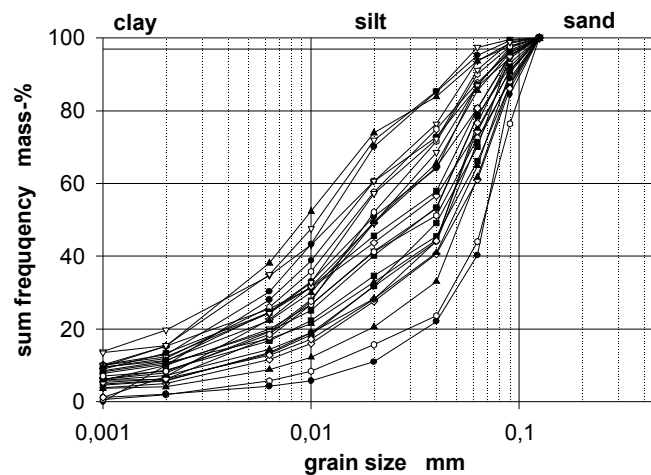
Correlating the actually measured  $^{137}\text{Cs}$  activity concentration of the sediment at all sampling sites to the average monthly discharge shows a medium to good result in the examined years between 1990 to 2001. This meets the original expectations, as a higher runoff would cause less fine sediment particles being deposited at the bottom of the river. The correlation coefficients in all three cases are around  $-0,5$ . Yet, if the corrected activity concentrations are correlated to the runoff, no significant relation can be detected, because the grain size distribution (Fig. 2), which is the single most important factor in sediment transport, is levelled out of the calculation.

### $^{137}\text{Cs}$ in suspended particles

Water samples, taken at the four different sampling sites on a monthly basis, have been first analysed regarding activity concentration of the suspended matter contained. In contrast to sediment samples no trend for a change in activity concentration can be identified, which is rather surprising as the activity concentration of soil transported into rivers is over the time decreased by physical decay as well as erosion.

Similar as for the sediment samples, a close relation between average monthly discharge and activity concentration was shown at all three sampling sites. The correlation

coefficients range from 0,5 in Wallsee, 0,54 in Greifenstein to 0,62 in Ottensheim. That clear influence of runoff on the activity concentration of suspended particles in water samples has been strongly expected, since a higher runoff induces increased sediment relocation. It, however, has to be noted, that the satisfactory correlation results largely depend upon high water events, while during normal water periods, only somewhat poorer correlation could have been established.



**Figure 2:** Sum frequency distributions of sediment samples, Danube, Austria, 2001

At the sites Ottensheim and Wallsee, the activity load by suspended matter, during normal as well as during high water periods, is clearly by the factor 2 higher than downstream at Greifenstein. This might be explained by lower flow velocities downstream, although sampling took place within power stations, which generally lower the flow velocity.

The highest activity concentration of water by radionuclides bound to suspended particles can be found during summer and the lowest during winter. That result seems to absolutely sensible, taking into account the increased discharge during summer and the reduced discharge during winter. High flow depth and high flow velocity during summer season cause more particles to be suspended and transported in the water. The lowered flowing depth and flowing velocity in winter prevent suspended particles in the water – the particles are deposited to the bottom.

### *<sup>137</sup>Cs dissolved in water*

After the analysis of the suspended matter in the water samples, the dissolved <sup>137</sup>Cs radionuclides have been analysed. As for suspended particles, no significant temporal trend of change in the activity concentration can be identified for dissolved <sup>137</sup>Cs. Concentrations in the range from 0,0005 Bq/l to 0,0020 Bq/l are found uniformly distributed at all three

sampling sites. Analysing the relation between average monthly runoff and activity concentration gives a good negative correlation upstream at Ottensheim, with a correlation coefficient of  $-0,59$ . At the sampling site of Wallsee, the correlation becomes less significant with a coefficient of  $-0,33$ . Downstream, at Greifenstein, no correlation can be shown (coefficient  $0,07$ ). It is, however, strongly recommended not to rely on those correlation upon further verification, as the results could be merely coincidence, especially when taking into account a missing valid explanation of the different qualities of correlation from upstream to downstream.

Activity concentrations during winter almost equal the ones during summer and the concentrations during spring match with those from autumn. Interestingly, the winter/summer concentrations are a bit below the spring/autumn concentrations.

The activity concentrations caused by dissolved matter show no correlation with the activity concentrations of the suspended particles.

Generally it can assumed, that the activity concentration remains constant over time as well as with different ambient conditions such as changing runoff.

## **Discussion**

The long monitoring period allows to make long-term estimations of the radionuclides in the river. The results show strong correlation between runoff and activity concentration of the sediment as well as suspended particles, which might imply a future possibility for a qualitative estimation of runoff, when no detailed river data are available.

Apart from a small number of surprising results, the monitoring effort rendered expected and comprehensible data, which could and should serve as a basis for future research in the field of river and sediment management as well as in the field of erosion control, if it will be possible to relate erosion mechanisms with the behaviour of radionuclides in the water.

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# COMPARISON OF PUBLIC EXPOSURES FROM DIFFERENT SOURCES OF RADIOACTIVE CONTAMINATION IN RECENT YEARS IN SLOVENIA

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## **Introduction**

In spite of that Slovenia is a small country it contains a considerable variety of radioactive sources, which cause radioactive contamination of the environment. These sources mostly belong to nuclear fuel cycle, as the Krško Nuclear Power Plant, the Žirovski vrh Uranium Mine (in the decommissioning), the TRIGA Research Reactor and Central low and intermediate level radioactive waste storage. Some other technological enhanced natural radiation sources, for example, the Šoštanj Thermal Power Plant have also an impact to the environment.

Discharges of radioactive materials to the environment from nuclear installations in Slovenia have been regularly controlled during all the periods of their operation and operational radiation monitoring has been performed as well. Monitoring started in early 1980s with the extensive programmes: at the Krško Nuclear Power Plant in 1981, in the Uranium mining and milling facility at Žirovski vrh (1985), at the TRIGA Research Reactor and at the Central low and intermediate level radioactive waste storage (1986). According to the new Law on radiation protection and nuclear safety from 2002 a control of discharges from other facilities (hospitals using radionuclides in nuclear medicine, other facilities with releases of natural radionuclides, etc) should be performed. All relevant data have been documented at the operators and sent to regulatory authorities [1].

The legal basis for control of discharges is set by the regulations. Only the discharge limits of the NPP have been defined by the competent authority according to the dose constraints of population exposure. The annual data on gaseous and liquid radioactive releases are used as input data for a dispersion model and further for radiation exposure assessment to the members of critical groups. All measurements of released radioactivity are performed by the operators themselves and supervised by the approved technical support organisations. Measurements of radioactivity at reference points are required where specific radionuclides under investigation are already present in the environment, especially some natural radionuclides. In such cases only the net contribution of environmental contamination is taken into account in dose assessment.

## **Regular radioactivity monitoring programmes**

### a) Global contamination in the environment

The basic scope of the radioactivity monitoring programmes was determined by the regulations. Radioactive contamination has been monitored in the following environmental media: in surface waters, air, soil (including external radiation), precipitation, as well as in



drinking water, foodstuffs and feedingstuffs. On the basis of average specific activities of radionuclides the exposure to a member of the public can be estimated. The effective dose to an adult, caused by the global contamination of the environment in the year 2002, was estimated to be 10.8 microSv/year [2]. The main contributors to the exposure are external radiation from deposited Cs-137 and ingestion of Sr-90 with local foodstuffs.

#### b) Nuclear Power Plant Krško

The regular monitoring at the Krško Nuclear Power Plant includes steady control of liquid and gaseous effluents at the discharge points and an independent off-site surveillance programme of the nearby surroundings. The off-site monitoring programme covers an area up to a radius of 12 km around the facility and comprises measurements of the radionuclide concentrations in the river water, sediments and fish, in air (particles and iodine), in precipitation, in soil samples, in underground and tap water, in food samples of vegetal and animal origin, and in feedingstuffs. External radiation is measured by TLDs and automatic gamma probes [3].

Conservatively estimated individual exposures for member of the public are based on the directly measured values in the environment and on model calculations. Main pathways which contribute to the dose are internal exposure through inhalation (H-3, C-14, I-131, particulates), ingestion (C-14) and external exposure through submersion and deposition (Xe-135 and Co-58, Co-60, Cs-137, respectively). The estimated effective dose for the member of the public in the year 2002 is 1,5 microSv/year and is well below the regulatory limit which is 50 microSv/year.

#### c) Research reactor and radioactive waste storage

The two nuclear installations operating at Brinje near Ljubljana are the TRIGA MARK II research reactor (250 kW) and the Central low and intermediate level radioactive waste storage.

Emissions and environmental concentrations are monitored for both nuclear facilities. Two exposure pathways were considered in dose assessment due to the reactor operation: external exposure due to Ar-41 immersion and ingestion of contaminated river water. The external immersion dose in the 2002 due to releases of Ar-41 into the atmosphere was estimated to be 0.29 microSv/year and a very conservative estimate of the ingestion dose due to release of radioactive effluents into the Sava river gives 0,05 microSv/year. A total annual effective dose of 0,34 microSv is received by a member of the public [4].

In the Central low and intermediate level radioactive waste storage at Brinje mainly disused sources from industry and medicine are stored. Radon releases and direct radiation from the storage are the main sources of public exposure and both were considered in radiological assessment. The annual dose received by a farmer occasionally working in a field nearby would be around 0.3 microSv [5].

#### d) Žirovski Vrh Uranium mine

At the former Uranium mine and mill at Žirovski Vrh relatively low grade ore was excavated and treated, containing less than 0.1% U<sub>3</sub>O<sub>8</sub>, in the period 1985-1990. Radioactive

wastes (0.6 million tonnes), such as chemical tailings and waste rock (1.5 million tonnes, 70 mg/kg  $U_3O_8$ ) are the main sources of the radioactive pollution to the environment.

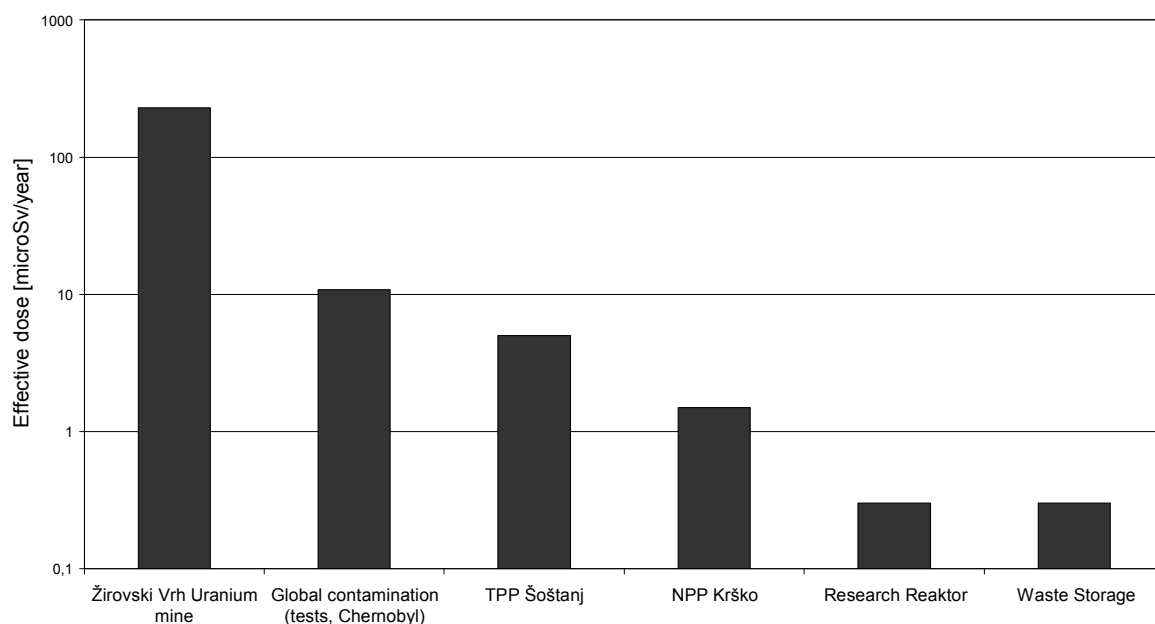
The regular monitoring programme of environmental radioactivity is not explicitly prescribed by the regulations and comprises measurements of long-lived natural radionuclides and radon in air, measurements of surface waters, sediments and water biota, food and grass and in the soil, as well as measurements of the external gamma radiation. Inhalation of short-lived radon progeny is the main contributor to the exposure with at least 75% additional exposure to the public [6]. Total annual effective dose for 2002 is estimated to be 240 microSv. Taking into account the total population exposure from natural radiation in this area (5.5 mSv), the former uranium mining at Žirovski Vrh contributed about 6% of this value. Due to a recent remedial works and reduction of radon exhalations in the last years the population exposure decreased considerably

#### e) Thermal power plants

Two coal-fired thermal power plants (Šoštanj, 750 MW and Trbovlje, 200 MW) produce fly ash with U-Ra contents of 350 and 180 Bq/kg, respectively. Measurements of stack emissions to the atmosphere in the Šoštanj TPP showed an enrichment of Pb-210 and Po-210 against Ra-226. Radioactive liquid discharges from fly ash disposal were reduced in 1994 by introducing a closed water cycle in the fly ash transport system. Currently contamination of environment (at nearby hilly sites) with Pb-210 due to stack emissions is under investigation. The environmental impact of fly ash disposal is mainly related to chemical water pollution. Some minor radiological impact could be observed (dose to population is of the order of magnitude of less than 5 microSv/year) [7].

Comparison of public exposures from different sources of radioactive contamination in the year 2002 is shown in Figure 1.

Figure 1: Annual exposures for members of the public from different sources of radioactive contamination.



## Conclusions

The comparison of the public exposure due to various sources of radioactive releases to the exposure of a members of the public in Slovenia shows that the critical group in the vicinity of the Žirovski Vrh uranium mine is the most exposed one in Slovenia. The global contamination due to the Chernobyl accident and the past nuclear tests was estimated to be around 10 microSv in Slovenia while the estimated annual dose for all other radioactive facilities are in the order of magnitude of one microSv. In this review the releases from the hospitals are not reported but some studies showed that it is not negligible.

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# AMBIENT DOSE EQUIVALENT $H^*(d)$ – AN APPROPRIATE PHILOSOPHY FOR RADIATION MONITORING ONBOARD AIRCRAFT AND IN SPACE?

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## **Introduction**

The large and during recent years steadily increasing number of radiation protection quantities may cause a confusing situation for the applicant. The complexity is reduced if it is recognized that our understanding of the biological effects of exposure to ionizing radiation is constantly developing and, thus, the functions, roles and definitions of quantities are modified. Quantities in the field of radiation protection have been introduced to meet two different needs apart from standardization: risk estimation and operational measurement. The “limiting” or “protection quantities” are defined by the International Commission on Radiological Protection (ICRP) in order to relate the radiation risk to a single dose quantity which takes account of the human as a receptor, of the different radiation sensitivities of various organs and tissues as well as the different radiation qualities [1]. Limiting quantities are not directly measurable and the International Commission on Radiation Units and Measurement (ICRU) has therefore developed “operational quantities”. It was agreed in ICRU Report 43 that a quantity suitable for metrology should be unified, *i.e.* the same for all radiation types, relatable to and failsafe for protection quantities, additive, defined at a point, relatable to an appropriate, though in some cases very simplified phantom of the human body, and directly measurable.

Harvey & Portal [2] gave an illustrative analogy featuring road safety to further clarify the complex situation in radiation protection with three classes of quantities. A complex parameter related to speed as well as vehicle and road conditions would certainly give a good indication of the risk of an accident, but would be completely unsuitable for routine application. The value of the quantity could not easily be linked by practical instrumentation

to legal standards of time, length and mass, apart from principle difficulties in measuring such a parameter. However, velocity can be measured with relatively simple devices and can, in principle, be made failsafe for the more complex parameter. The concession one has to make is that the situation probably becomes over-safe when conditions are good. The pictorial analogy to radiation protection is the following: operational quantities shall serve as a conservative and directly measurable estimate of protection quantities by which limits are stated. We will see later that this requirement is not always fulfilled in practical applications.

### **Definition of the Ambient Dose Equivalent $H^*(d)$**

For the purposes of routine radiation protection, it is desirable to characterize the potential irradiation of individuals in terms of a single dose equivalent quantity that would exist in a phantom approximating the human body. The phantom selected is the so-called ICRU sphere for made of 30-cm-diameter tissue-equivalent plastics with a density of  $1 \text{ g/cm}^3$  and a mass composition of 76.2 % oxygen, 11.1 % carbon, 10.1 % hydrogen and 2.6 % nitrogen. The “ambient dose equivalent”,  $H^*(d)$ , at a point in a radiation field is the dose equivalent that would be produced by the corresponding expanded and aligned field at a depth  $d$  in the ICRU sphere, on the radius opposing the direction of the aligned field. In an expanded field the fluence and its directional and energy distribution have the same values throughout the volume of interest as in the actual field at the point of reference. An expanded and aligned radiation field requires additionally that the fluence is unidirectional. For strongly penetrating radiations a reference depth,  $d$ , of 10 mm was recommended.

### **Application of $H^*(10)$ onboard Aircraft?**

Our analysis on the applicability of the  $H^*(10)$ -concept to in-flight dosimetry has to include two steps. Firstly, we need to investigate the appropriateness of dose equivalent in complexly mixed radiation fields. Secondly, the selection of the reference depth,  $d$ , of 10 mm must be discussed. Afterwards we shall be able to answer the question if the philosophy of  $H^*(d)$  is a reliable method to estimate risk-related whole-body protection quantities.

The definition of dose equivalent is based on the quality factor,  $Q$ , which is related to the relative biological effectiveness (RBE) of the radiation of interest. As the radiation climate onboard aircraft is composed of several particle types and energies, the assessment of the quality factor becomes extremely complex and would, in principle, imply the determination of the energy and LET spectra of all radiations involved. It is immediately clear that this task cannot be achieved in routine dosimetry.

Apart from metrological difficulties, it is questionable if the quality factor sufficiently considers the different radiation qualities in order to ensure adequate radiation protection. Major fractions of the dose accumulated in-flight are deposited by neutrons and protons in a broad energy range of several orders of magnitude [3]. Interaction processes within the human body lead to a significant modification of the primary particle spectrum. Depending on the energy of the incident particles, the reaction products include charged particles and photons. If  $H^*(10)$  is used, the quality factor is determined in a depth of 10 mm and the alteration of the spectral composition is thereby neglected.

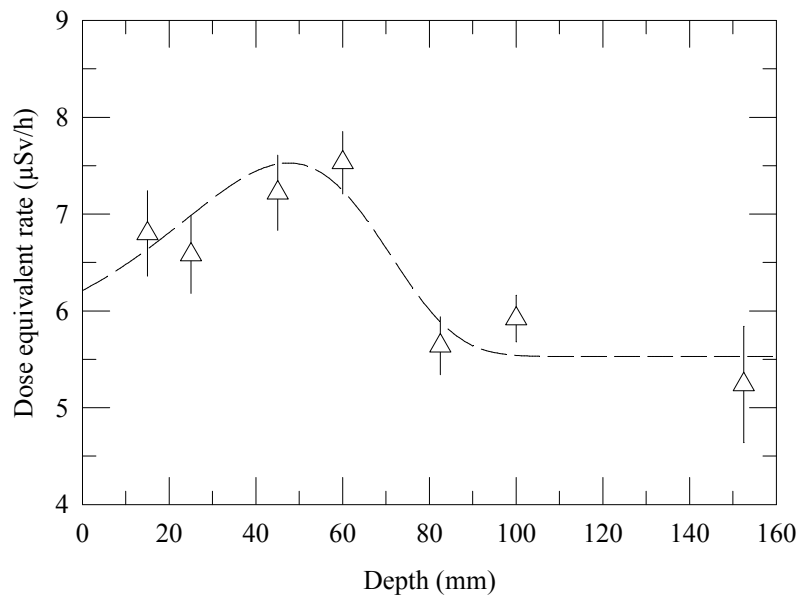


Figure 1. Depth distribution of dose equivalent measured onboard a series of eight return flights Cologne-Washington, DC.

A reference depth of 10 mm was recommended for strongly penetrating radiation, as the ICRU believed that the maximum dose equivalent occurs in this depth. To verify this assumption and measure the depth distribution of the dose equivalent onboard aircraft, we used polyethylene spheres of different diameters equipped with small thermoluminescent detectors (TLDs) of the commercial type TLD-700 ( $^7\text{LiF: Mg, Ti}$ ) in their centres. The TLDs were evaluated according to the High-Temperature Ratio (HTR) Method [4-5] developed at the Atomic Institute. This method utilizes the well-investigated relative intensity of the combined high-temperature glow peaks 6 and 7 compared with the dominant peak 5 [3] as an indication of the dose-average LET and from it calculates the quality factor of a mixed radiation field of unknown composition. Figure 1 shows the depth dose distribution recorded onboard a series of eight trans-atlantic flights between Cologne and Washington [6]. It is

clearly visible that the maximum dose equivalent appears in a depth of 50 to 60 mm rather than 10 mm, implying that  $H^*(10)$  underestimates the actual whole-body radiation exposure.

For the space radiation environment the skin dose equivalent appears to be a conservative estimate of the whole-body effective dose. Figure 2 displays the dose equivalent profile measured in different depths of a water-filled phantom body onboard the Russian Space Station Mir [7].

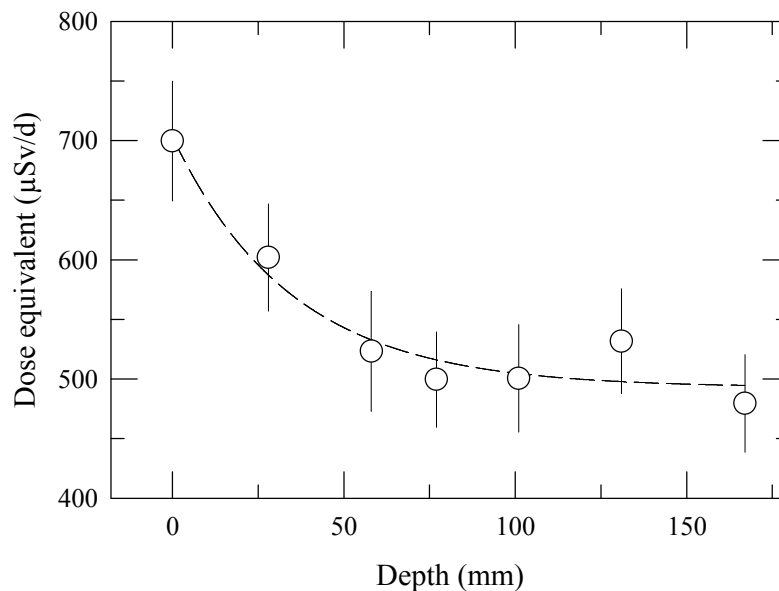


Figure 2. Depth distribution of dose equivalent measured onboard Space Station Mir perpendicular to the spacecraft hull.

The discussion and the carried out experiments demonstrated that the philosophy of  $H^*(10)$  leads to an underestimation of the whole-body radiation exposure when applied onboard aircraft and in space. It therefore has to be considered to introduce a new concept that could be based on microdosimetric principles, offering the unique potential of a more direct correlation to radiobiological parameters.

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# A PASSIVE BONNER SPHERE SPECTROMETER FOR IN-FLIGHT NEUTRON SPECTROMETRY

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## **Introduction**

The radiation field at typical civil aviation altitudes of about 10 km above sea level involves a broad spectrum of particles and energies. Because of their high biological effectiveness, expressed by an energy-dependent quality or radiation weighting factor, neutrons play a dominant role in dose assessment. However, neutrons are effective not only in causing biological hazards, but may as well affect aircraft electronics containing several gigabytes of semiconductor memory by producing so-called single event effects (SEE), *e.g.* bit-flips. The precise determination of the spectral distribution of the neutron fluence rate – shortly termed the neutron energy spectrum – onboard aircraft contributes to a better understanding of the radiation load on aircrew personnel and frequent flyers and may as well stimulate the improvement of reliability and availability of both appropriate shielding materials and avionics hardware. Metrological difficulties concern chiefly the energy of the cosmic-ray induced neutrons extending from meV to some GeV and the discrimination of charged particles in the detector signal.

## **Instrumentation and Methodology**

The Bonner Sphere Spectrometer (BSS) [1] is currently the only device providing a sufficient energy response over several orders of magnitude up to some hundred MeV. The system consists of a detector for thermal neutrons located in the centres of moderating spheres of different diameters. In successive elastic collisions neutrons lose kinetic energy. According to the diameter of a specific sphere, a fraction of neutrons with “appropriate” primary energies gets moderated until thermal equilibrium with the atoms of the surrounding medium is reached. The neutron energy spectrum needs to be unfolded from the count rates of the complete multisphere set by means of sophisticated mathematical techniques, *e.g.* the



maximum entropy method or iterative recursion algorithms – in practice assisted by the computing power of modern microprocessors.

The commonly employed thermal neutron sensors are active, *i.e.* power-consuming devices offering the advantage of a generally high counting efficiency and real-time neutron detection. However, the reading of active detectors, such as  $^3\text{He}$ ,  $^6\text{Li}(\text{Eu})$  or  $\text{BF}_3$  counters, in a complexly mixed radiation environment is biased significantly by charged particle-induced reactions in the detector volume that cannot be rejected by simple amplitude discrimination. Figure 1 shows as an example the pulse height spectrum obtained from a  $^6\text{Li}(\text{Eu})$  scintillation counter in different radiation fields. The clearly visible full-energy peak of the  $^6\text{Li}(n_{\text{th}}, \alpha)^3\text{H}$  reaction is overlapped by an exponentially decreasing proton signal. If the detector is operated in counting mode – as in most practical applications – the protons are falsely identified as neutrons and adulterate the unfolded spectrum. In contrast, the discrimination of gamma-rays, depositing their energy in the detector volume via secondary electrons, is unproblematic.

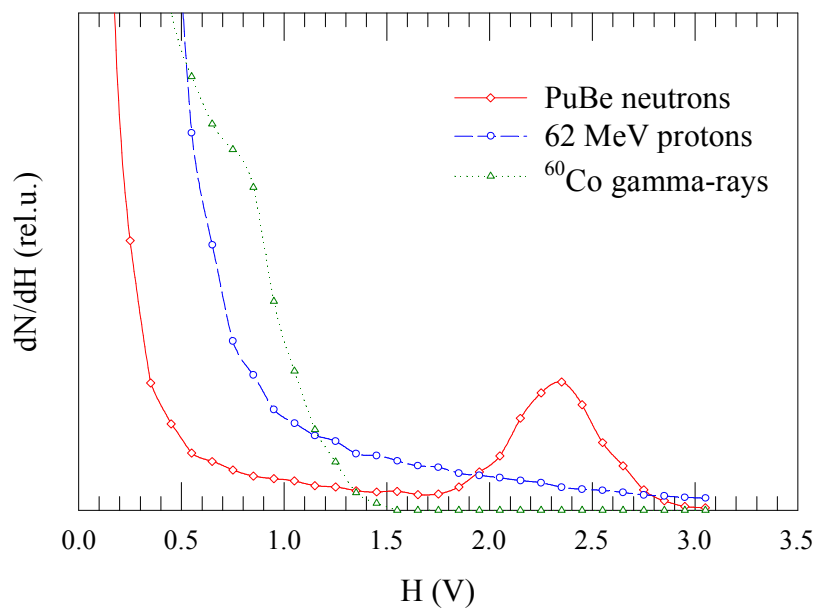


Figure 1. Pulse height spectrum obtained from a  $^6\text{Li}(\text{Eu})$  scintillation counter in different radiation fields.

The limitation of active devices can be overcome by the so-called Pair Method with passive  $\text{LiF}:\text{Mg}, \text{Ti}$  thermoluminescent dosimeters (TLDs) of different types. TLD-600 ( $^6\text{LiF}$ ) and TLD-700 ( $^7\text{LiF}$ ) detectors show almost identical responses to photons and charged particles, but very different thermal (and epithermal) neutron efficiencies. This property stems from the  $^6\text{Li}(n_{\text{th}}, \alpha)^3\text{H}$  reaction dominating the TLD-600 neutron response. The reaction cross section is 943.2 barn for 0.0253 eV neutron energy, compared with a total neutron cross

section value of 14.7 barn for  ${}^7\text{Li}$  and the same energy. The thermal neutron-induced TL signal may be ascertained by subtraction of the TLD-700 from the TLD-600 TL glow curve. By arranging TLD-600 and TLD-700 dosimeters in pair, the net thermal neutron dose may therefore be determined in mixed radiation fields without the influence of charged particles.

### Passive Bonner Sphere Spectrometer

The passive Bonner sphere spectrometer of the Atomic Institute consists of seven high-density ( $0.953 \text{ g/cm}^3$ ) polyethylene spheres with diameters of 2,  $3\frac{1}{2}$ , 5,  $6\frac{1}{2}$ , 8, 10 and 12 inches and a bare detector. Six TLD-700 chips are arranged as a cube in the centre of the each sphere, surrounded by six TLD-600 chips. The TLDs are read out with the laboratory-made TL-DAT.II system. The neutron energy spectrum is unfolded from the count rates by means of the MAXED algorithm applying the maximum entropy principle [2]. The passive BSS was calibrated in the beams of PuBe and AmBe isotopic neutron sources, in the neutron beam of our TRIGA Mark-II research reactor as well as in the CERN-EU High-Energy Reference Field (CERF) [3] which closely resembles the cosmic-ray induced neutron environment in the high atmosphere. In Figure 2 the spectra recorded in the CERF are compared with a FLUKA Monte-Carlo simulation [4-5]. All spectra are presented as fluence per unit lethargy, meaning that the area under the curve in a certain energy interval is directly proportional to the neutron fluence in that interval.

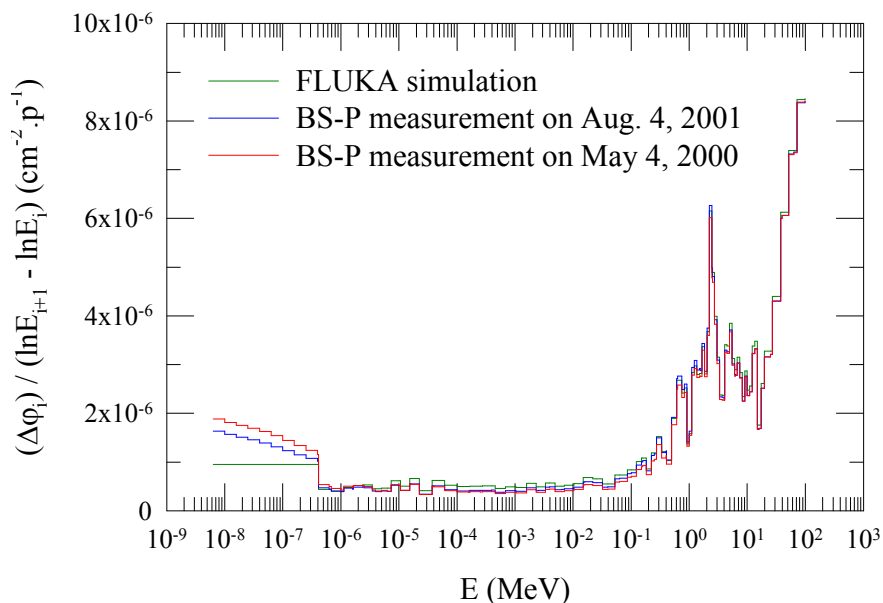


Figure 2. Neutron energy spectrum measured in the CERF and compared with a FLUKA Monte-Carlo simulation.

## In-Flight Neutron Spectrometry

The neutron energy spectrum was measured onboard a series of four return flights between Cologne (Germany) and Washington, DC (USA) during the solar minimum conditions of June/July 1996. This experiment was world-wide the first determination of the neutron spectrum by passive spectrometry methods onboard an aircraft. The project was conducted with the support of the German Airforce which reserved two seat rows in the front of an Airbus A 310-304 for the passive BSS. The count rates were computed according to the Pair Method and unfolded with MAXED. Figure 3 gives the resulting spectrum in comparison with a FLUKA simulation for 10.9 km altitude [6] which was the average altitude of the flights. The spectrum shows two relative maxima around 1 and 100 MeV, the second of which cannot be completely assessed by the applied BSS. The agreement between measurement and calculation proved to be excellent.

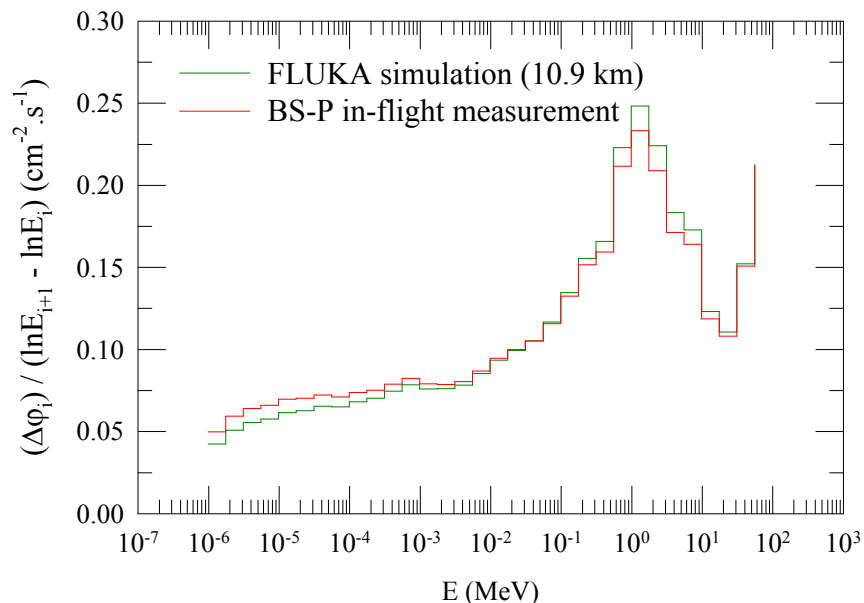


Figure 3. Neutron energy spectrum measured in-flight and compared with a FLUKA Monte-Carlo simulation.

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# **Determination of effective dose in anisotropic gamma radiation fields: application of dosimeters calibrated in terms of Hp(10)**

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Radiation Protection Institute ATS Ukraine

# Problems with determination of E for external exposure

- Protection quantity E cannot be measured
- Operational quantity Hp(10) strictly cannot be measured too
- Hp(10) may be approximated by readings of properly designed and calibrated personal dosimeter

# ICRU definition of $H_p(d)$

The **personal dose equivalent**,  $H_p(d)$ , is the dose equivalent in soft tissue, at an appropriate depth,  $d$ , below a specified point on the body

# IAEA Safety Guide RS-G-1.3

3.30. For radiation protection purposes the measured operational quantities  $H_p(10)$  and  $H_p(0.07)$  are interpreted in terms of the protection quantities effective dose  $E$  and equivalent dose to the skin and extremities  $H_T$ . To do this, realistic assumptions have to be made with respect to the type and uniformity of the radiation field and the orientation of the worker within the field.

*Under these conditions, the dosimeter reading gives a good estimate of the worker's exposure without underestimating or severely overestimating the relevant protection quantity.*

# Established data on conversion coefficients

ICRP 74:

Conversion coefficients from air kerma free-in-air  $K_a$  to organ doses  $D_T$  and  $E$

Five standard geometries (AP, PA, LAT, ROT, ISO)

Energies from 10 keV to 10 MeV



# Limitations of ICRP 74

- No detailed angular dependences
- No conversion from  $H_p(10)$  to E

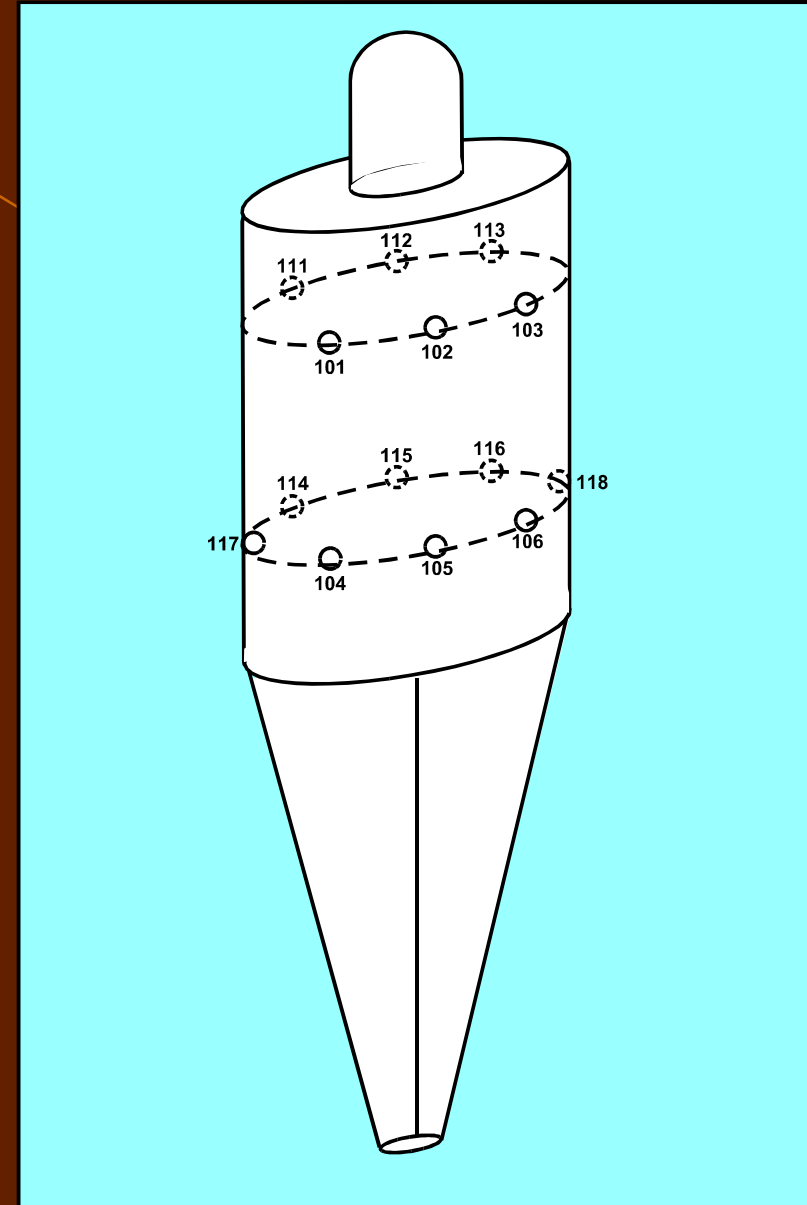
# Other relevant publications

- Kim et al (1999) – detailed angular dependences for  $E/K_{\alpha}$  conversion coefficients for 80 keV, 300 keV and 1 MeV
- Zankl (1999) – conversion from strictly defined  $Hp(10)$  to  $E$  for standard geometries and energy from 10 keV to 10 MeV

# Description of the problem

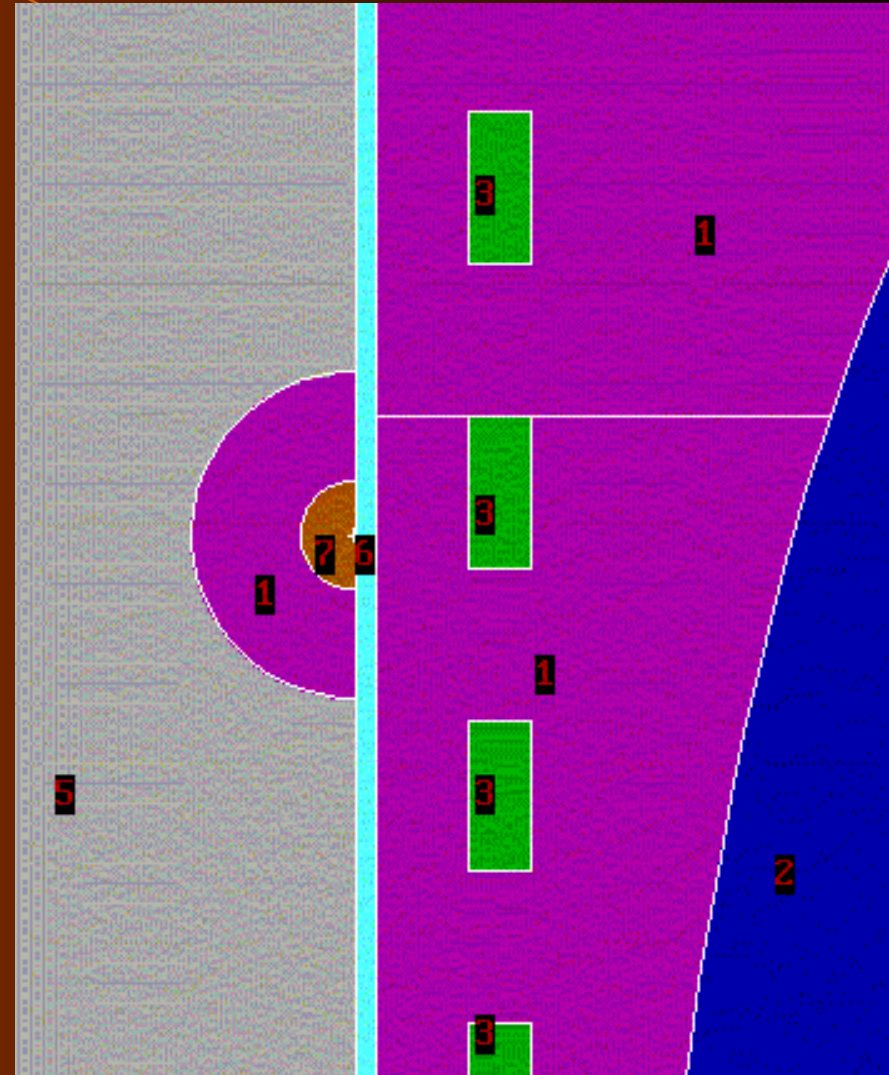
- Monte Carlo (MCNP-4B) simulation of organ doses  $D_T$ ,  $E$  and  $Hp(10)$  for a number of dosimeter placements
- Parallel beams with incidence angles in  $4\pi$  geometry with  $\pi/6$  increment over polar and azimuth angles
- Energies 50, 80, 100, 200, 300, 600 keV and 1 MeV

# ADAM phantom and placement of dosimeters



# Dosimeter design

- 1 – soft tissue,
- 2 – lung tissue,
- 3 – bone tissue,
- 5 – air,
- 6 – skin tissue,
- 7 – LiF.



Partial conversion coefficient for  
dosimeter  $k$ :  
particular energy  $\varepsilon$  and incidence  
angle  $\Omega$

$$C_k(e, \Omega) = \frac{E(e, \Omega)}{Hp_k(10, e, \Omega)}$$

Integral conversion coefficient for  
dosimeter  $k$ :  
composite source including  
different energies and directions

$$C_k = \frac{\sum_{i,j} w_T H_T(e_i, \Omega_j)}{\sum_{i,j} H_{p_k}(10, e_i, \Omega_j)}$$

# Benchmark calculations

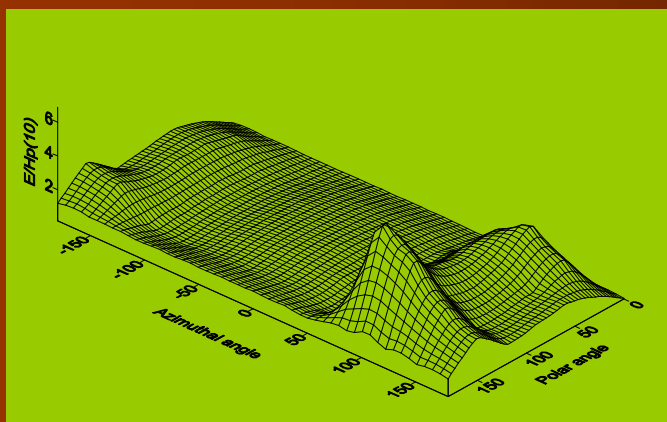
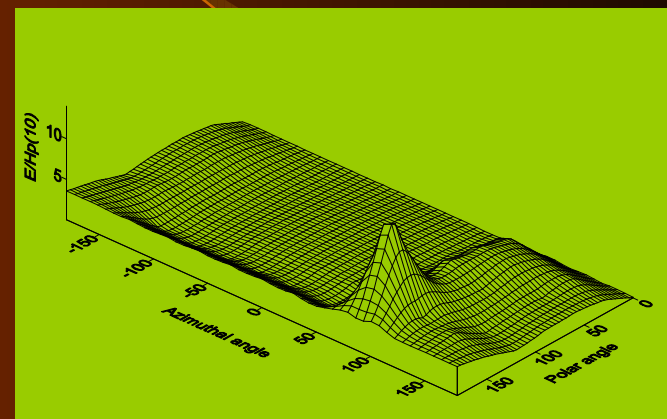
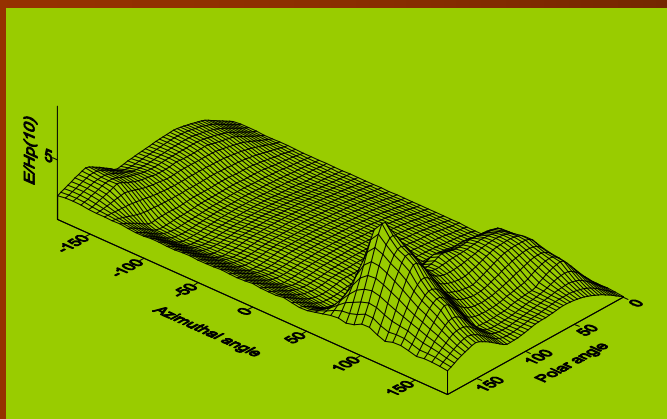
Effective dose ( $E$ ) per unit air kerma ( $Ka$ ) free-in-air,  $E/Ka$  (Sv Gy<sup>-1</sup>).

Sources of information: A – ICRP 74, 1996, B - Kim et al, 1999, C - this work.

Photon energy		80 keV			300 keV			600 keV		
Azimuthal angle		0°	60°	90°	0°	60°	90°	0°	60°	90°
Polar angle	Source of data									
	A	n/a	n/a	n/a	n/a	n/a	n/a	n/a	n/a	n/a
30°	B	0.77	0.49	0.33	0.66	0.45	0.30	n/a	n/a	n/a
	C	0.79	0.45	0.23	0.73	0.46	0.25	0.75	0.53	0.31
	A	n/a	n/a	n/a	n/a	n/a	n/a	n/a	n/a	n/a
60°	B	1.30	0.86	0.55	0.99	0.72	0.49	n/a	n/a	n/a
	C	1.28	0.84	0.43	1.09	0.76	0.45	0.98	0.79	0.53
	A	1.43	n/a	0.55	1.09	n/a	0.56	1.02	n/a	0.60
90°	B	1.47	0.98	0.61	1.08	0.82	0.54	n/a	n/a	n/a
	C	1.44	0.95	0.47	1.18	0.86	0.50	1.04	0.85	0.57



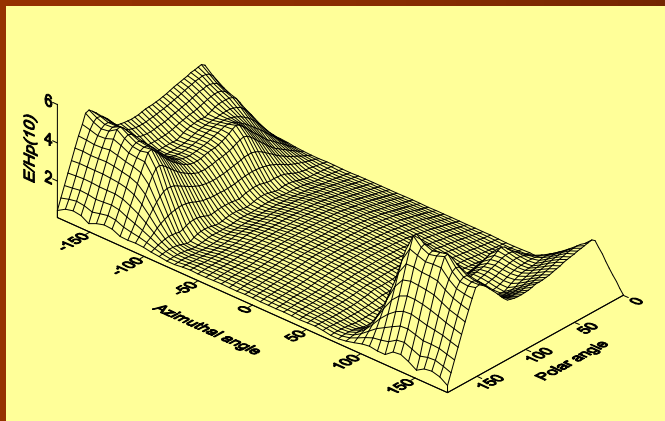
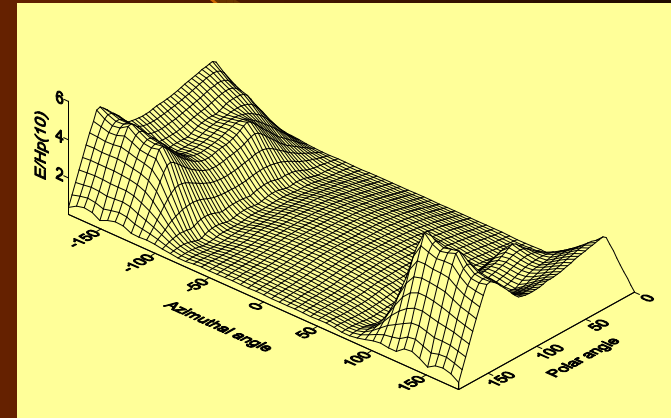
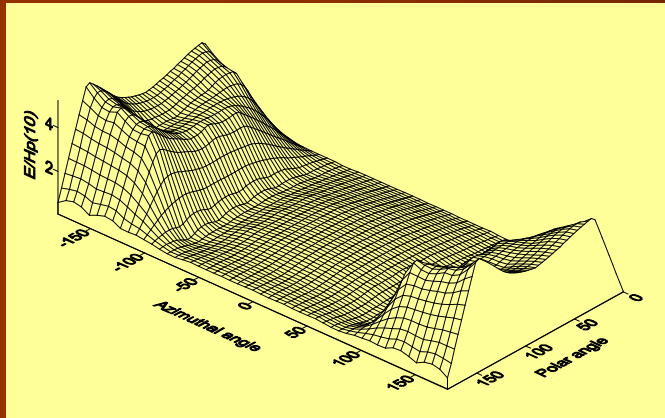
# Partial $E/H_p(10)$ conversion coefficients for **left** chest dosimeter



80 keV    600 keV

200 keV

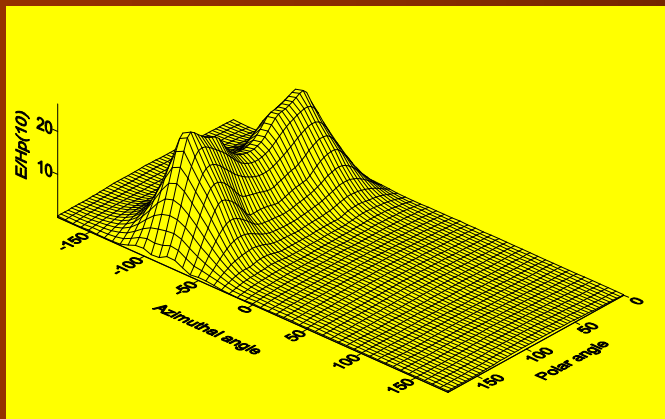
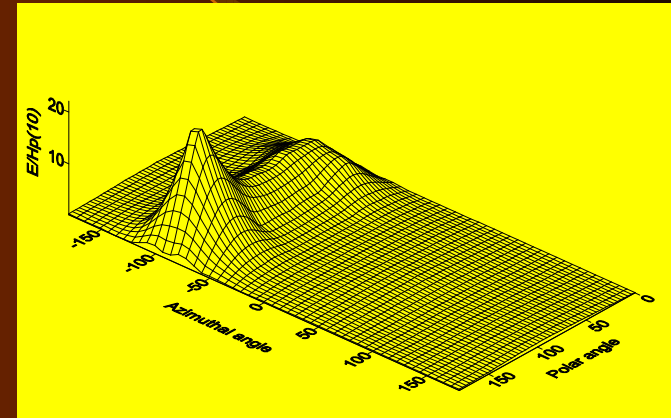
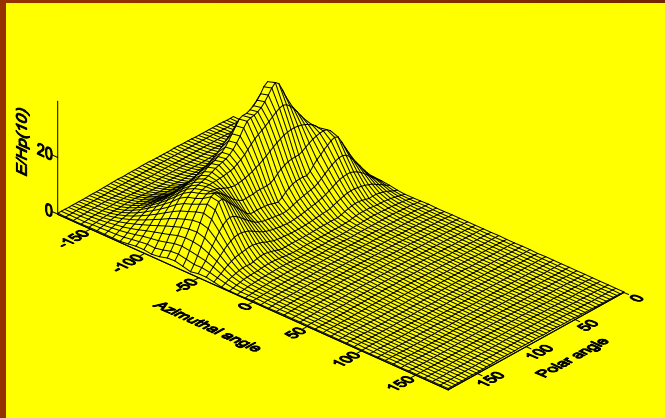
# Partial $E/H_p(10)$ conversion coefficients for **central** chest dosimeter



80 keV    600 keV

200 keV

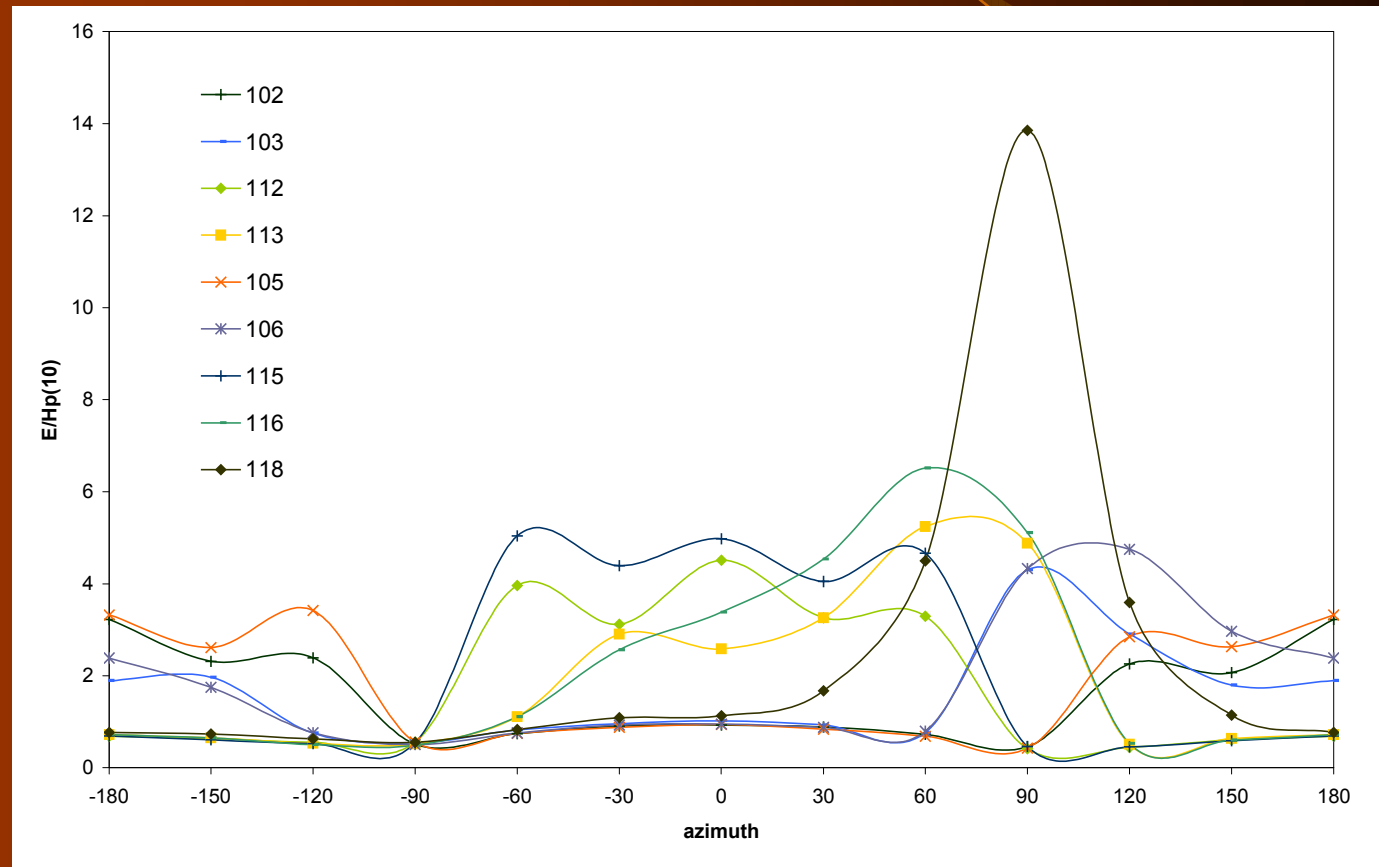
# Partial $E/H_p(10)$ conversion coefficients for right hand dosimeter



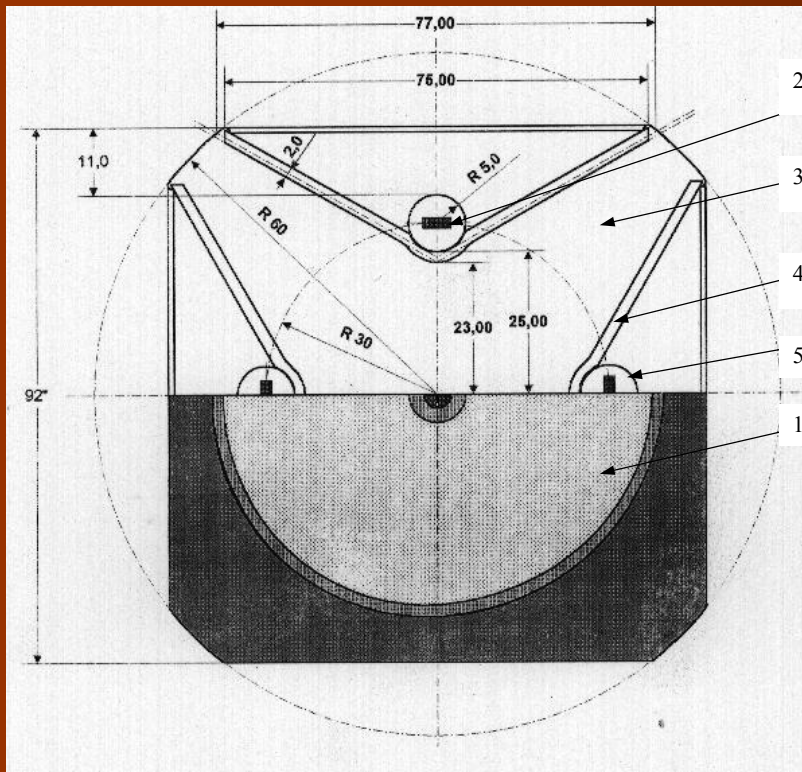
80 keV    600 keV

200 keV

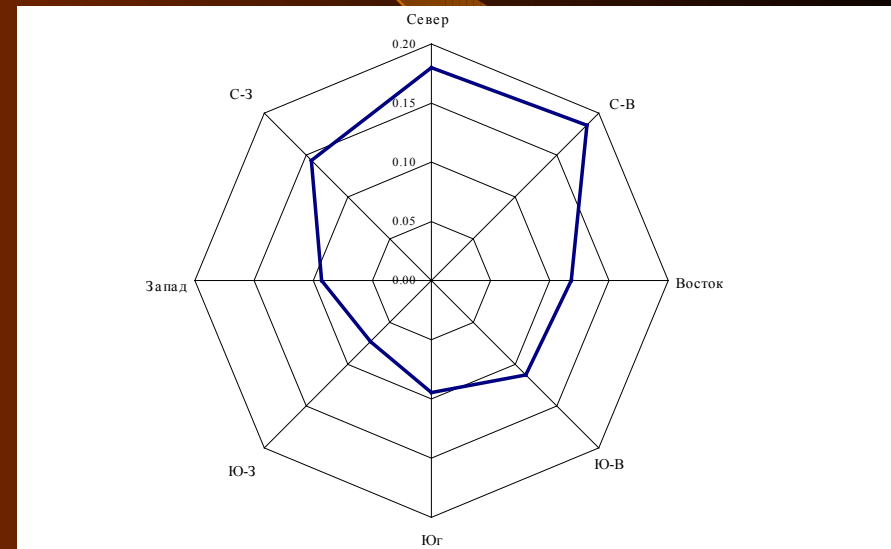
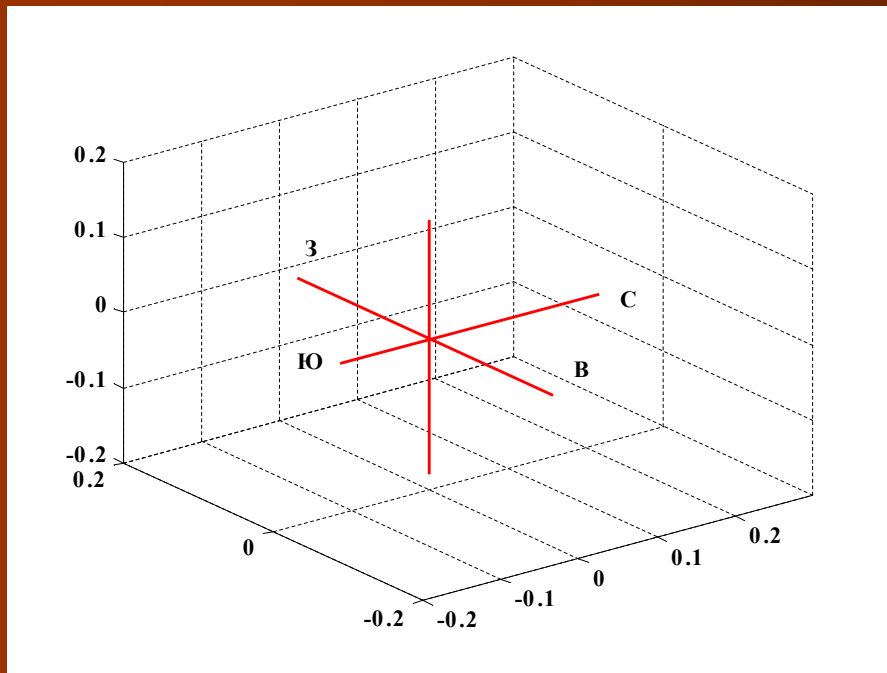
# Dependence of $E/Hp(10)$ behavior on the placement of dosimeter: parallel beam (incidence normal to the axis of the phantom, 200 keV)



# Determination of angular distribution of photon field



# Anisotropy of photon field at Shelter (room Г347)



# Measure of anisotropy

$$h = \frac{\max(I_k) |_{k=1,6}}{\min(I_k) |_{k=1,6}}$$

# Classification of the degree of anisotropy

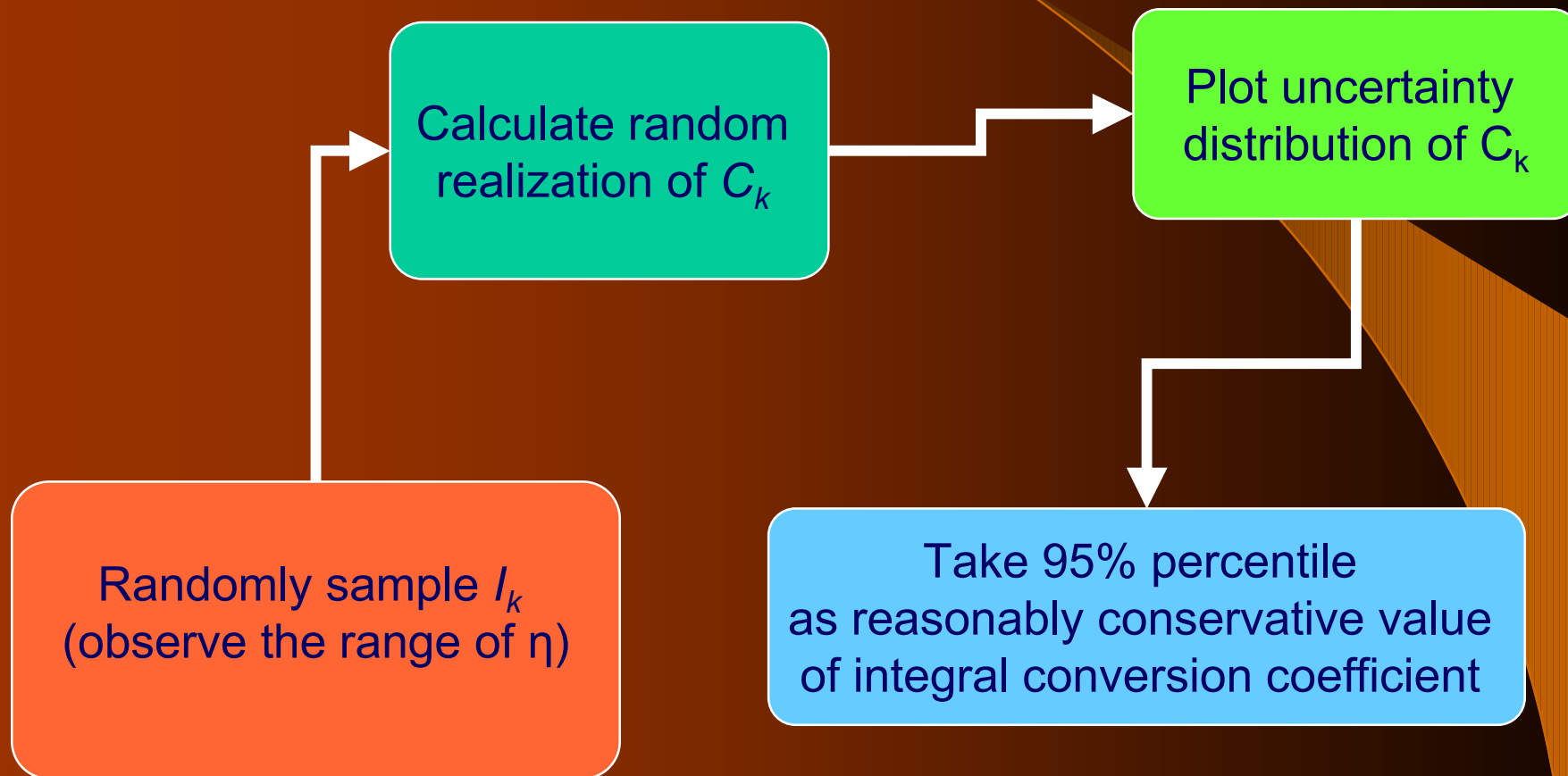
- Isotropic:  $\eta < 1.4$
- Quasi-isotropic:  $1.4 < \eta \leq 2$
- Anisotropic:  $\eta > 2$



# Assessment of integral conversion coefficient

1. Stochastic simulation of workplace field with given energy spectrum and degree of anisotropy
2. Constraint:  $E'$  (estimate of  $E$ ) should not underestimate the actual  $E$  in 95% of cases

# Flow-chart for evaluation of integral conversion coefficient



# Assessment of $E$ using overall conversion coefficient

$$E' = C_k(e, h) H p_k (10)$$

# Conversion from $H_p(10)$ to $E$ for conditions of “Shelter” and left chest dosimeter

- Isotropic ( $\eta < 1.4$ ) - 0.98
- Quasi-isotropic ( $1.4 < \eta \leq 2$ ) - 1.0
- Anisotropic ( $\eta > 2$ ) -

*multiple dosimeter approach*

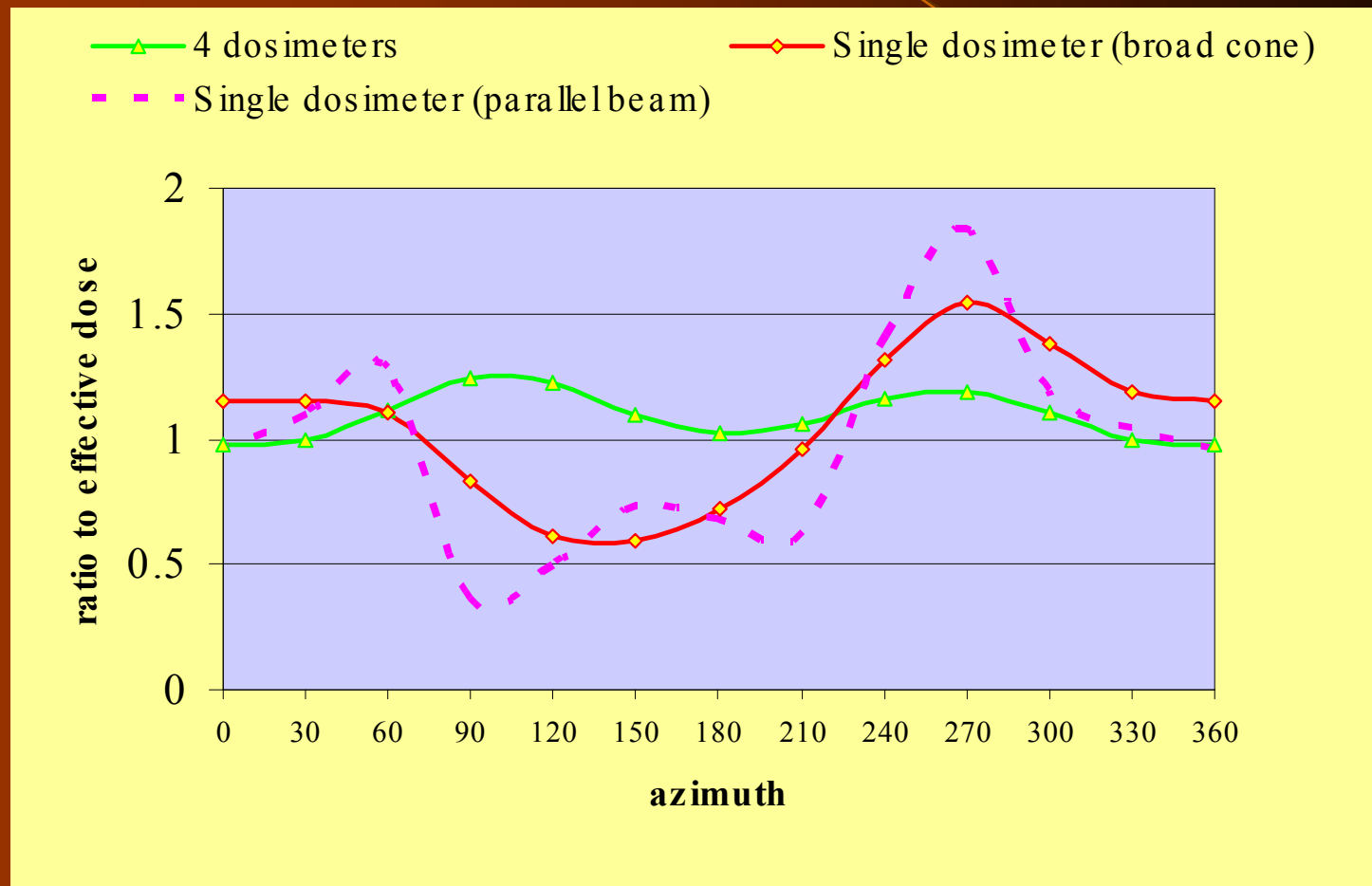
# Multiple dosimeter approach: assumptions

- $E' \geq E$
- $E' = \sum \alpha_k H p_k(10)$
- $\sum \alpha_k = 1$

# Multiple dosimeter approach: results

- $\alpha_{front\ left}$  = 0.35
- $\alpha_{front\ right}$  = 0.35
- $\alpha_{rear\ left}$  = 0.15
- $\alpha_{rear\ right}$  = 0.15

# Assessment of $E$ by single dosimeter and multiple dosimeters



# Conclusions

1. Straightforward application of  $H_p(10)$  as surrogate for  $E$  may not work under certain conditions
2. Partial data on behavior of  $E$  and  $H_p(10)$  for different dosimeters allow to estimate  $E/H_p(10)$  conversion coefficients for any particular composite source



# Conclusions (continued)

3. In practical situations, anisotropy of workplace fields may be measured by six- collimator device assessing contribution to a dose from six orthogonal directions
4. Reasonably conservative conversion coefficients may be assessed for given energy spectrum and degree of anisotropy of workplace fields

# Conclusions (continued)

5. For strongly anisotropic fields multiple dosimetry approach gives the best estimate of  $E$  comparing to plain  $H_p(10)$  readouts or integral conversion coefficients

# Neutron Effective Dose calculation behind Concrete Shielding of Charge Particle Accelerators with Energy up to 100 MeV

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## **Introduction**

All accelerators of the charged particles with energy up to 100 MeV represent a complex geometrical design with set of sources of radiation. The most penetrating component of radiation at the working accelerator is the neutrons of a wide energy spectrum. The shielding should provide not excess of the required dose limits. Calculations of shielding and the doses of neutrons behind shield are used various methods and, created on their basis, algorithms.

Method Monte-Carlo allows carrying out direct modeling of radiation transport through substance in real geometry. The programs, created on its basis, require essential expenses of computer time at account of a dose behind rather thick shields. Much more practicable method is phenomenological approach based on existent experimental and calculated data of dose attenuation by shielding.

The purpose of work is the comparison of results of calculations of effective neutron doses behind concrete shielding by a method Monte-Carlo and by phenomenological method.

## **The initial data and geometry of calculation**

Geometry of barrier shielding (see fig. 1) was chosen for comparison of calculation methods. The thickness of protection varied from 25 cm up to 300 cm with a step 25 cm. The calculations have been performed by Monte Carlo and phenomenological methods for monoenergetic neutrons with energy from 5 MeV to 100 MeV as well as for neutron spectra produced by protons with energies of 30 MeV and 72 MeV in thick targets/1, 2/.

The calculation by a method of Monte-Carlo was carried out with the program MCNP4B /3/ (package of libraries DLC189). The geometry of calculation is shown in a fig. 1. The statistical error of the calculated data with method of Monte-Carlo did not exceed 5%.

Over a fairly wide range of shield thickness it has been found/4/ sufficiently accurate to estimate shielding with simple equations of the form:

$$E_{ef}(d, E) = \frac{f}{R^2} h(E) B(E) \exp(-d / I(E)) \quad (1)$$

This equation applies to a source of monoenergetic neutrons of energy  $E$  with symbols having the following meanings:

$E_{ef}(d, E)$  is the effective dose behind shield with thickness of  $d$ ;

$f$  is neutron yield per steradian;

$R$  is the distance from the source to the point of interest outside the shield;

$h(E)$  is the conversion coefficient that relates neutron fluence to the effective dose/5/;

$B(E)$  is the build-up factor of neutrons /4/;

$I(E)$  is the attenuation length for effective dose through the shield/4/.

This formula considers the energy dependence of neutron dose attenuation length with shield thickness by adopting the  $\lambda$  and the  $B$  values after deep penetration in a spectral equilibrium state. The phenomenological method based on expression (1) was used.

## Results

The results of calculations are shown in a fig. 2-8. In a fig. 2 the results of calculation of attenuation of neutron effective dose by concrete shield obtained with use of two described above methods are submitted. Factor of attenuation of a dose  $k$  is defined as the relation:

$$k = E_{ef}(d, E) / E_{ef}(0, E), \quad (2)$$

where  $E_{ef}(0, E)$  - effective dose of neutrons in a point located on distance  $R$  from a source at absence of shield.

The neutron effective dose as a function of incident neutron energy calculated for concrete shield of 1 m, 2m and 3 m in thickness is shown in Fig 3. Figures 4, 5 and 6 show the result of calculations of effective dose as a function of a concrete shield thickness when the shield irradiated by neutrons produced in a copper thick target by protons with energies 72 MeV and 30 MeV/1, 2/.

Attenuation length  $I(E)$  and build-up factor  $B(E)$  were calculated with MCNP code (Monte Carlo method) and compared with the same parameters used by phenomenological method/4/. Figures 7 and 8 show comparison of these parameters.

## Conclusions.

Data obtained by these two methods agree within factor 2 over considered range of neutron energies and shielding thickness. Comparison of the results shows that difference in shield thickness between calculated by Monte Carlo and phenomenological method is not exceeded half-value layer for neutron effective dose, that is from 10 cm to 30 cm for considered energies and thickness of shields.

### References

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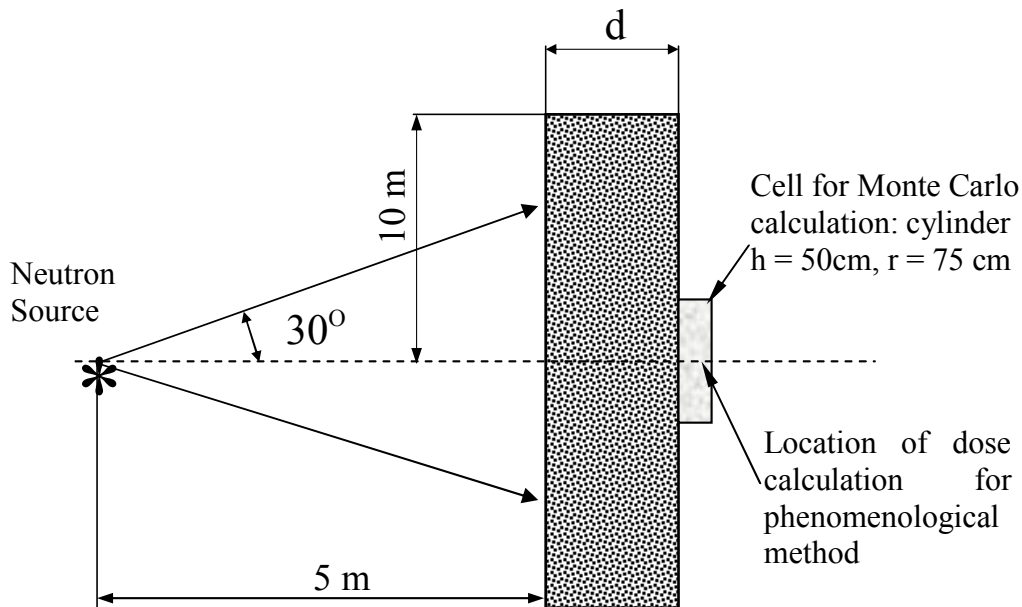


Fig.1. Geometry of shield for comparison of calculation methods. The thickness of the shield is varied from 25 cm up to 300 cm with a step of 25 cm.

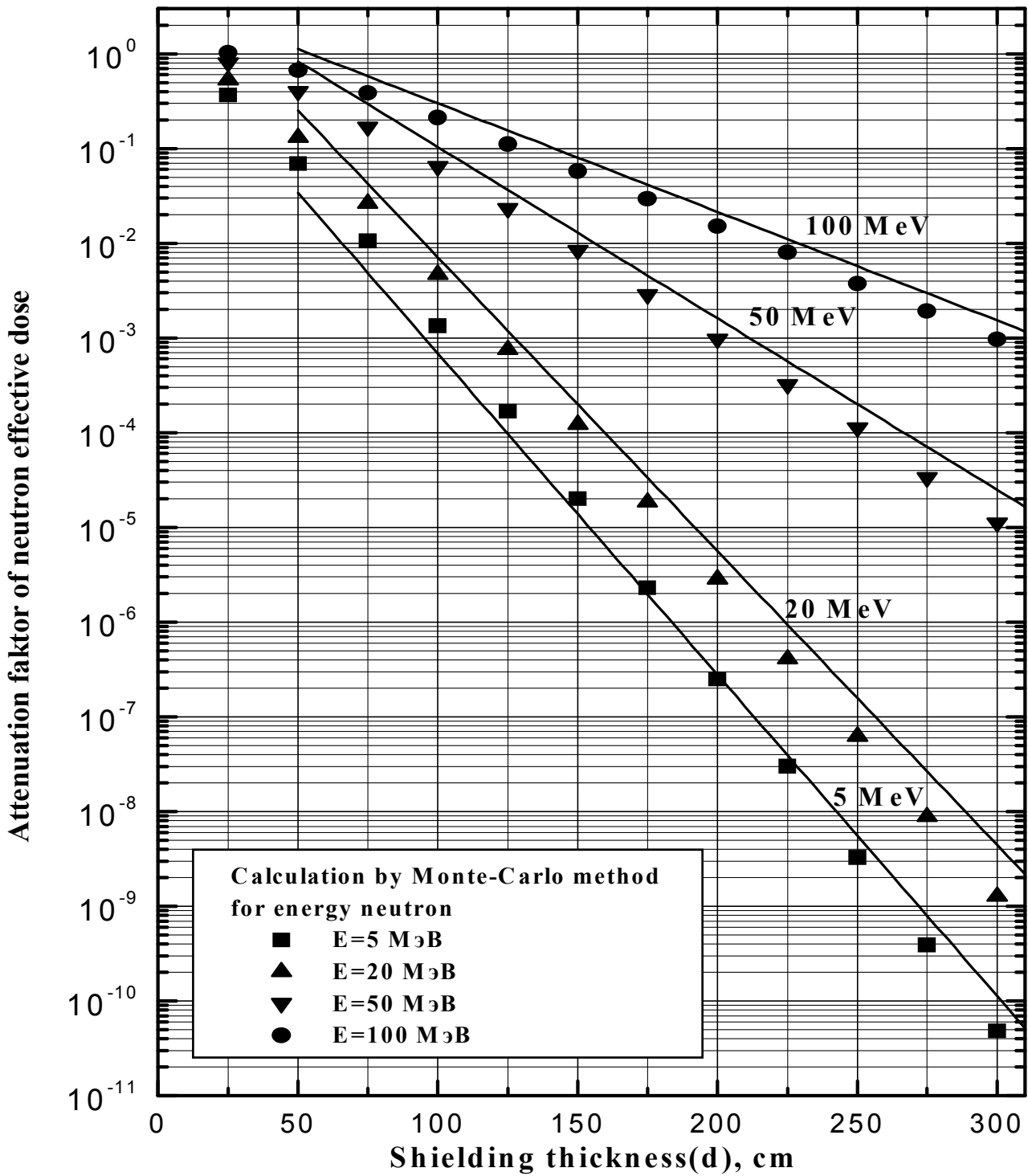


Рис. 2. Comparison between Monte Carlo and phenomenological methods calculated data of neutron effective dose attenuation by concrete. Isotropic point neutron source with energy 5, 20, 50 and 100 MeV at 5 m from shield. ■, ▼, ▲, ● – Monte Carlo calculation, curves – phenomenological method.

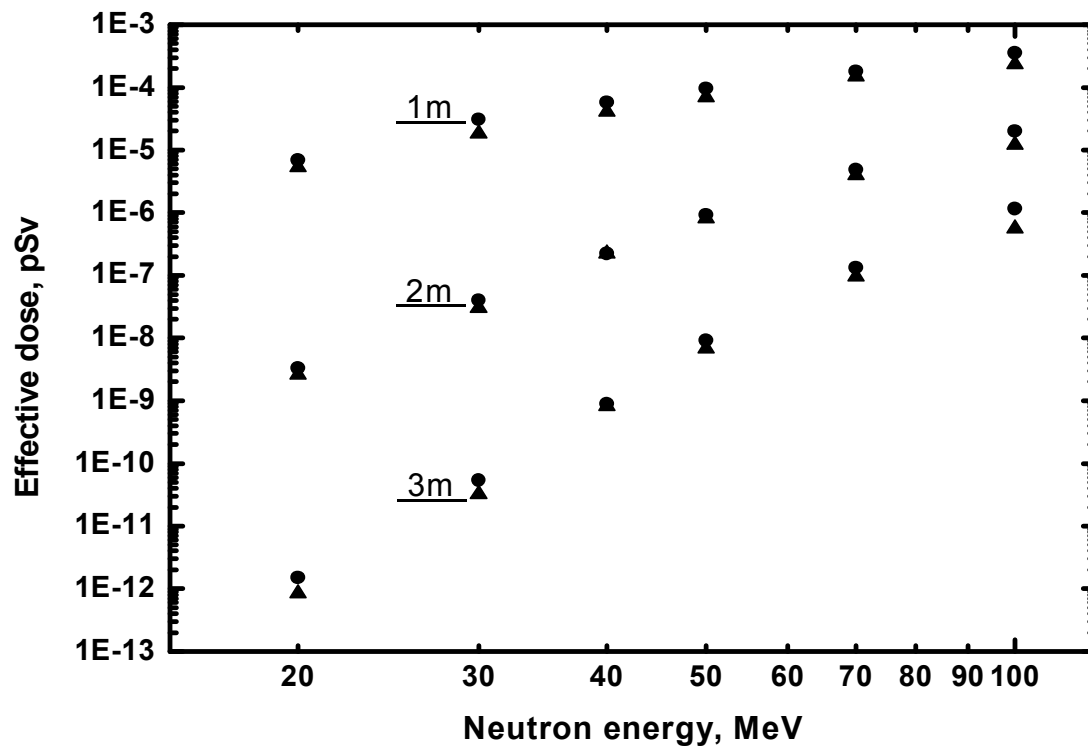


Рис. 3. The neutron effective dose as a function of incident neutron energy calculated for concrete shield of the thickness indicated. ● - phenomenological method, ▲ - Monte-Carlo

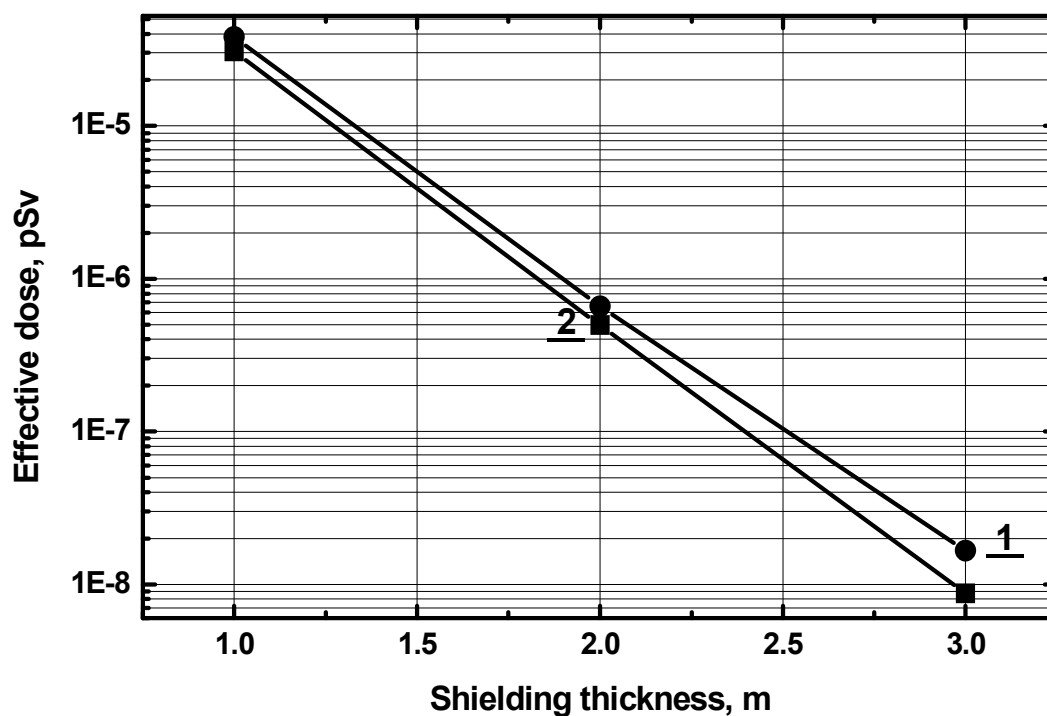


Fig. 4. Comparison of effective dose calculations as a function of a concrete shield thickness when the shield irradiated by neutrons in a copper thick target by protons with energies of 72 MeV produced at an angle of  $9^0$  with an axis of proton beam:  
1 – phenomenological method, 2 - Monte-Carlo

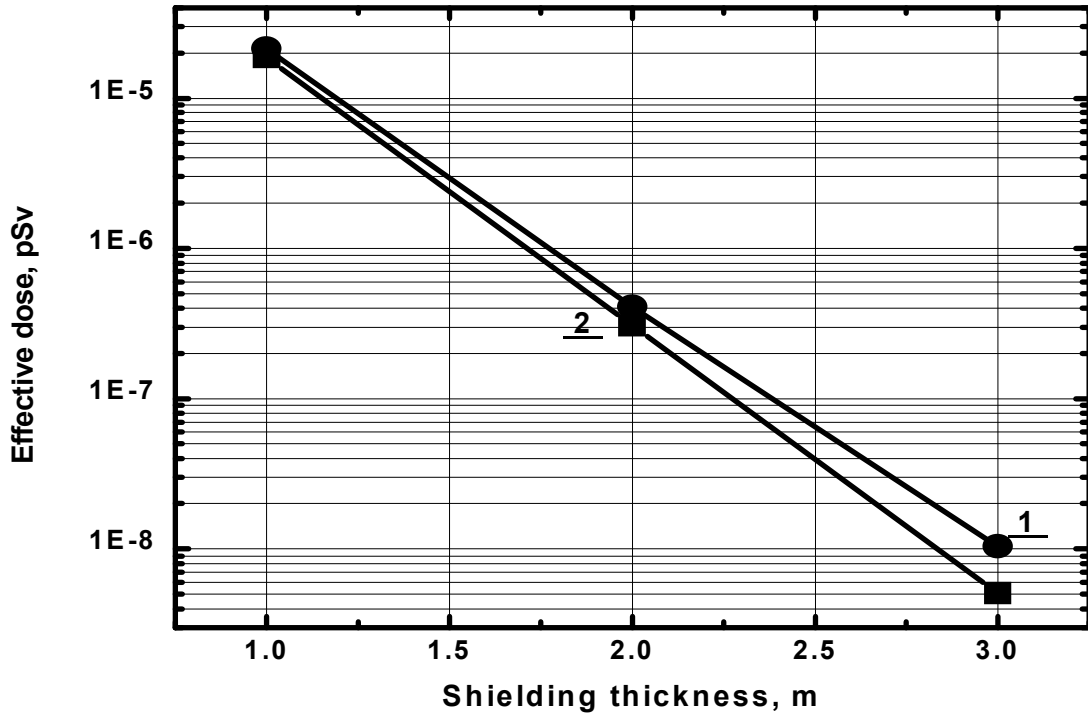


Fig. 5. Comparison of effective dose calculations as a function of a concrete shield thickness when the shield irradiated by neutrons produced in a copper thick target by protons with energies of 72 MeV at an angle of  $90^0$  with an axis of proton beam:  
 1 – phenomenological method, 2 - Monte-Carlo

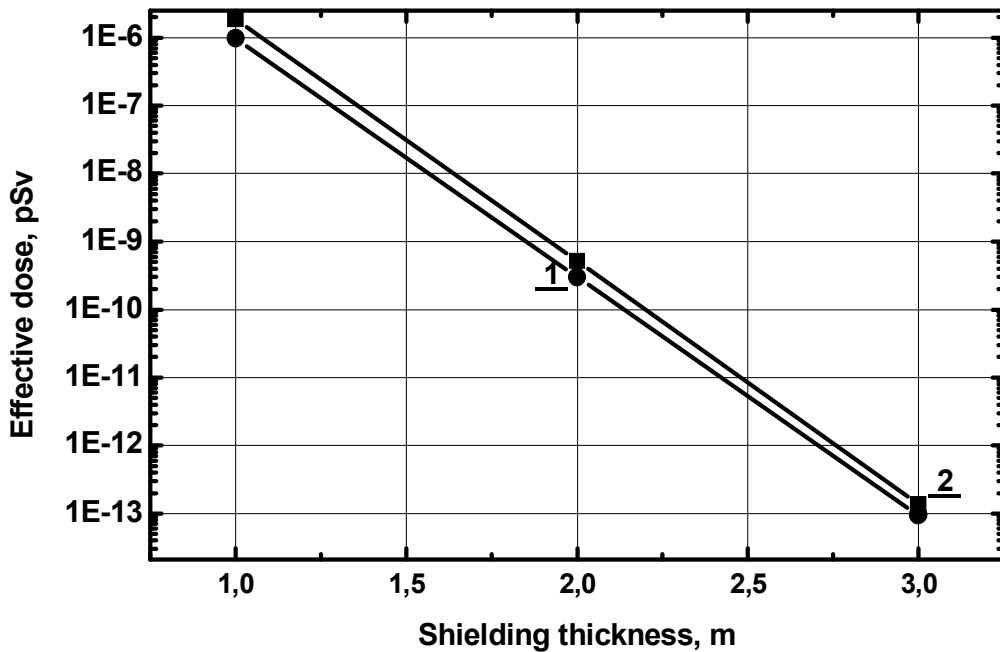


Fig. 6. Comparison of effective dose calculations as a function of a concrete shield thickness when the shield irradiated by neutrons produced in a copper thick target by protons with energies of 30 MeV at an angle of  $0^0$  with an axis of proton beam:  
 1 – phenomenological method, 2 - Monte-Carlo



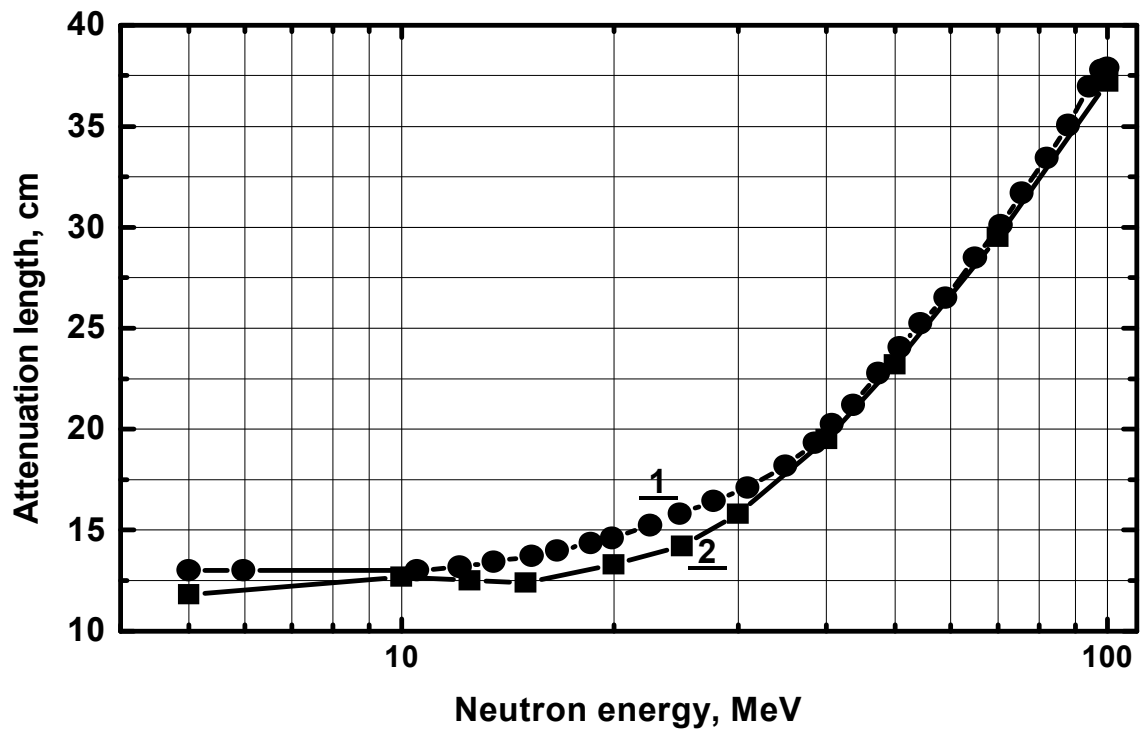


Fig. 7. The variation of the attenuation length of effective dose for monoenergetic neutrons in concrete as a function of neutron energy.  
 1 – phenomenological method, 2 - Monte-Carlo

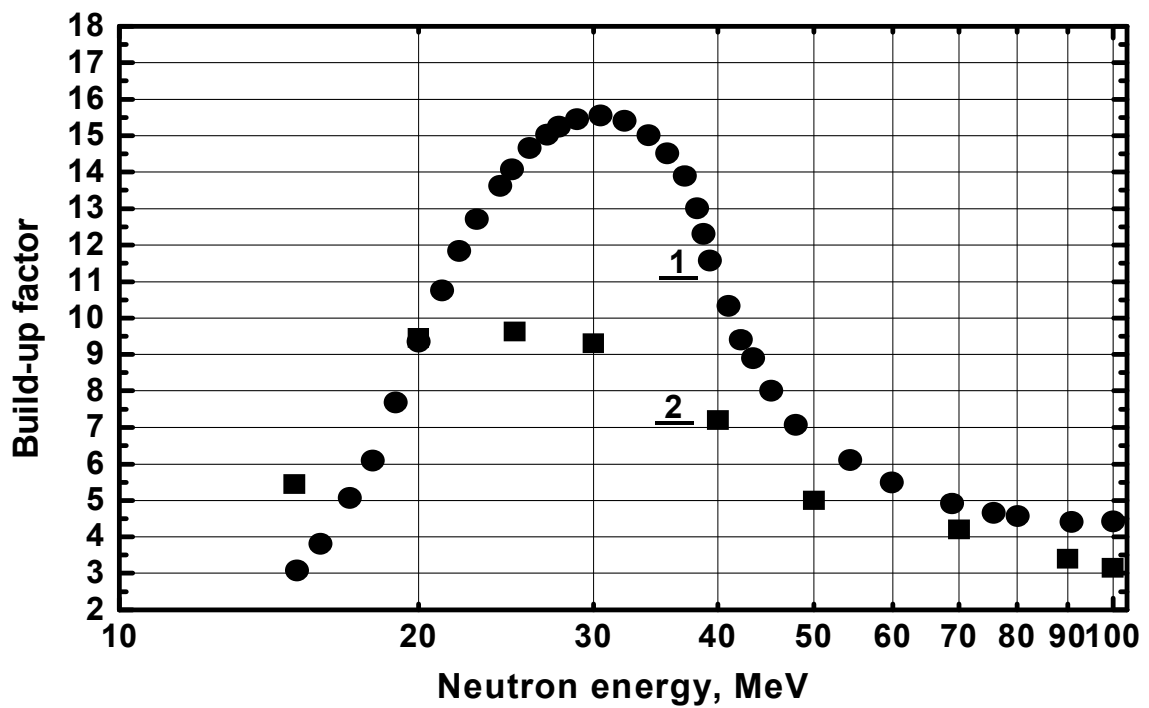


Рис. 8. Build-up factor for effective dose of neutrons:  
 1 – phenomenological method, 2 - Monte-Carlo

# COMBINING TLD AND ACTIVATED CHARCOAL DETECTORS FOR RADON DOSIMETRY

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## **Introduction**

Because of its large adsorption ability, charcoal is often used to collect radon. The principle of the charcoal method is based on the measurement of the gamma emission of radon from the charcoal by means of gamma rays detectors such as sodium iodide (NaI(Tl)). Some of those detectors are equipped with a diffusion barrier between the charcoal and the air, therefore the radon adsorbed is proportional to the radon concentration in the air [1-3].

The possibility to combine the track-etched detector with charcoal to increasing the detector sensitivity was shown in the last years [4-5]. Special requests are necessary in the case of a personal dosimeter when the detector must be active only during working activity [5].

TL dosimeters have only a limited utilization in radon dosimetry because they are generally sensible to whole spectrum of natural radiation: cosmic, terrestrial or building material radioactivity. But they can be used for radon progenies (RP) measurement. The TLD type contains an air-sampling pump that draws a continuous, uniform flow of air through a detector assembly. The detector assembly includes a filter and at least two TLDs. One TLD measures the radiation emitted from radon decay products collected on the filter, and the other TLD is used for a background gamma correction. This RP is intended for a sampling period of 48 hours to a few weeks. Interpretation of the results of this measurement requires a calibration for the detector and the analysis system based on exposures to known concentrations of radon decay products [6].

Stranden [7] shown that charcoal detectors can be equipped with a TLD chip inserted into the charcoal, the main radon concentration is then determined using a TLD

reader. The main characteristics involved in the design of a combined “Charcoal –TLD” dosimeter are: the alpha background radioactivity of the charcoal, the adsorption and desorption time of radon and water from charcoal directly connected with thickness of charcoal bed, the  $k$  value of adsorption coefficient and the breakpoint of the charcoal which is directly related with the water mass from the charcoal. Recently, Scarpitta, [8] was done a comparative study using the best charcoal, Calgon PCB, and a new charcoal sort, Carboxen 564. It was found that the last have superior characteristics regarding the adsorption properties of the radon especially in thin layers. Very recently studies about a TLD-Charcoal dosimeter were made by a Polish team [9] which was used a very sensitive TLD chip namely MCP-N (LiF:Mg,Cu,P) produced in the own laboratory.

### **Experimental Method**

By placing TLD crystals inside an open charcoal container, the beta and gamma radiation from radon daughters, produced by the decay of adsorbed radon, is detected continuously during the exposure time.

The adsorption constant, also the gamma background radioactivity of the different charcoals was measured by gamma spectrometry using a Ge calibrated detector. For the radon evidence the main two photo peaks of the radon progeny (607 keV and 352 keV) were used. Radon concentration in the air was measured using a Radim device. The sample Nr. 5 (Ro-2F, from Table 1) was measured in Ghent [5] and remeasured in our laboratory. It is a Romanian sort of charcoal (CASS-Buzau) and this sort was used in our experiments. For low radon concentration the charcoal radioactivity itself can contribute to the TLD signal. A special device for rapid and complete charcoal cleaning (complete activation) was used. The Polish TLD chips namely MCP-N (LiF:Mg,Cu,P) type, 3 mm in diameter and 1.5 mm thickness were installed at the center of two kinds of small charcoal plastic canisters of 10 ml and 30 ml in volume. The canisters were exposed typically for 120 h in a radon chamber (58 l) at four different concentrations: 850 Bq/m<sup>3</sup>; 1240 Bq/m<sup>3</sup>; 1970 Bq/m<sup>3</sup> and respectively 3260 Bq/m<sup>3</sup>. The radon concentration was continuously measured with the Radim monitor device inside. After exposure, the reading of TLD was made with Harshaw -2000 device.

### **Results and discussion**

In Table 1 it can see the background radioactivity of some charcoals measured with a Ge high resolution spectrometer after a very good degassing and kept in old air for two weeks. Carboxen 564 have a very low background closed of Ge background whereas our

charcoal Ro-2F has the background with 10% higher, especially due to radium and cesium content. Also from this table we can see that adsorption capacity of Carboxen 564 is about five times higher as our used charcoal-Ro-2F.

Table 1. Gamma background and adsorption constant (k) for different charcoals.

Type	$E_g > 90 \text{ keV}$		Photopeak area		k	Obs .
	(A)	(B)	(609 keV)	(352 keV)		
CalgonPCB	-	107976	48	92	3.61	$^{137}\text{Cs}, ^{238}\text{U}, ^{40}\text{K}$
George	101380	100800	49	52	2.75	background
Norit R1	112600	109263	180	250	2.9	$^{226}\text{Ra}, ^{137}\text{Cs}$
Ro-1G	121280	117500	233	346	4.02	much $^{226}\text{Ra}$
<b>Ro-2F</b>	<b>109700</b>	<b>110400</b>	<b>161</b>	<b>201</b>	<b>1.42</b>	$^{137}\text{Cs}, ^{226}\text{Ra}$
Merck	-	104200	-	-	1.05	low activity
PZ	-	108300	-	-	1.52	$^{137}\text{Cs}$
Carboxen	100700	100600	58	80	7.01	low activity
<b>Ge backgr.</b>	<b>100700</b>	<b>100600</b>	<b>30</b>	<b>54</b>	-	-

The results for 4 TLD chips simultaneous exposed in radon chamber at  $1240 \text{ Bq/m}^3$  (5 days) in  $V_1$  (30 ml) and  $V_2$  (10 ml) canisters are shown in Table 2. The TLD intensity of background was for these chips  $B=2\eta C$ . As can see, the reproducibility is better as 10%. If we consider in  $V_1$  case the detection limit as  $B^{1/2}=1.41 \eta C$  the minimum detectable concentration for one day exposure is about  $200 \text{ Bq/m}^3$ . The result recently obtained by Bogacz *et al.* [9] is  $100 \text{ Bq/m}^3$  for three days of exposure. Considering usage of Carboxen 564 the limit of detection can be five times smaller, therefore  $40 \text{ Bq/m}^3/\text{day}$ , rather small and promising to be used for a radon personal dosimeter at workplaces.

Another experiment was the exposure of 3 TLD+Charcoal and others 3 TLD only chips in the laboratory room (about  $60 \text{ Bq/m}^3$ ) for 5 days. The average corrected background signal was two times higher in the case of TLD+ charcoal sets.

In Fig.1 are presented the results obtained for exposure TLD+Charcoal in the radon chamber at four different concentrations also for five days. A linear dependence between the signal intensity and radon content in the radon chamber can be seen both for  $V_1$  and  $V_2$  sample types as in paper [9]. Although  $V_1$  canister is three times larger as  $V_2$  it can see from Table 2 and Fig.1 that the intensity of the signal is only with 27% higher. Probably the optimum canister volume is about 40-45 ml therefore about 20-25 g of charcoal.

Table 2. Four TLD+Charcoal exposures at  $1240 \text{ Bq/m}^3$  for 5 days

Chip number	$P_1$	$P_2$	$P_3$	$P_4$	Mean	St. deviation
TLD signal " $V_1$ "( $\eta C$ )	33.7	32	29.1	29.7	31.16	2.12
" $V_2$ "( $\eta C$ )	24.1	27.2	25.3	21.7	24.57	2.29

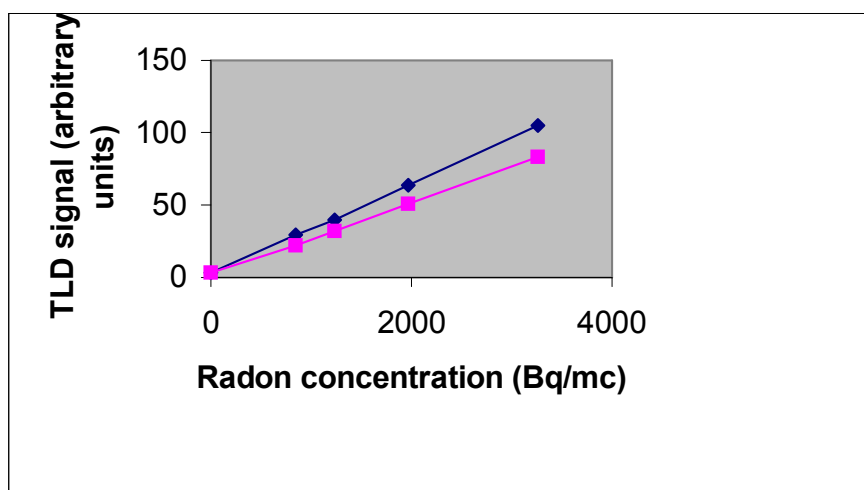


Fig.1. Signal emission intensity of TLD+Charcoal versus the radon concentration

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# DOSIMETRIC CHARACTERISTICS OF LITHIUM BORATE SOLID TL DETECTORS WITH DIFFERENT DOPANTS

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## **Introduction**

Although the lithium borate thermoluminescent (TL) detectors have been known for more than 30 years, significant improvements were made over the past several years by the selection of appropriate activators and the preparation of several new types of these detectors in the form of sintered pellets in the Institute of Nuclear Sciences Vinča [1-4]. In this work the characteristics of  $\text{Li}_2\text{B}_4\text{O}_7:\text{Cu,In}$ ,  $\text{Li}_2\text{B}_4\text{O}_7:\text{Cu,In,Ag}$  and  $\text{Li}_2\text{B}_4\text{O}_7:\text{Cu,Ag,P}$  are compared with the characteristics of the well known type,  $\text{LiF}:\text{Mg,Ti}$  detector (TLD-100). The following characteristics were investigated: the glow curve properties, batch homogeneity, relative sensitivity, detection threshold, fading, linearity and reproducibility of the response.

Special attention was given to the energy dependence of the detectors for medium and low energy X-rays. The energy dependence was determined in terms of “kerma in air” and in terms of  $H_p(10)$  (on the ISO phantom) according to the ISO Standard [5] requirements for TL detectors and TLD systems, respectively.

## **Materials and Methods**

The properties of  $\text{Li}_2\text{B}_4\text{O}_7:\text{Cu,In}$ ,  $\text{Li}_2\text{B}_4\text{O}_7:\text{Cu,In,Ag}$  and TLD-100 as well as the parameters used for the readout and pre-irradiation annealing are presented in [6]. For  $\text{Li}_2\text{B}_4\text{O}_7:\text{Cu,Ag,P}$  identical parameters were used as in [6] for  $\text{Li}_2\text{B}_4\text{O}_7:\text{Cu,In,Ag}$ .

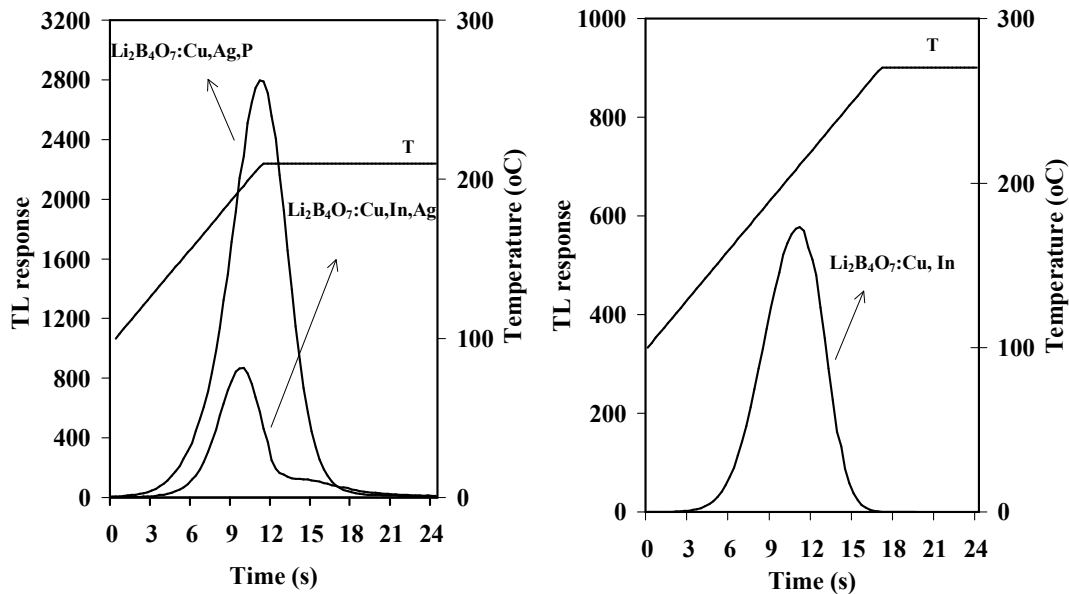
Details about the method of energy dependence determination for irradiations “free in air” and “on the phantom” were described in our previously published papers [6,7].

## **Results and Discussion**

### *Glow curves*

It was reported that the glow curves for all three types of lithium borate TL detectors consist of the two well separated TL peaks, the main dosimetric peak and the low temperature

peak. The main dosimetric peaks were at about 170°C for  $\text{Li}_2\text{B}_4\text{O}_7:\text{Cu,In,Ag}$  and at 210°C for two other types,  $\text{Li}_2\text{B}_4\text{O}_7:\text{Cu,In}$ , and  $\text{Li}_2\text{B}_4\text{O}_7:\text{Cu,Ag,P}$ . The low-temperature peaks are found at about 125°C for  $\text{Li}_2\text{B}_4\text{O}_7:\text{Cu,In,Ag}$  and  $\text{Li}_2\text{B}_4\text{O}_7:\text{Cu,In}$  [3] and at 135°C for  $\text{Li}_2\text{B}_4\text{O}_7:\text{Cu,Ag,P}$  [4]. Taking the reading at least one day after irradiation, as well as using the preheat treatment (100°C for 6 s) in this work, resulted in the elimination of the unstable low temperature peaks as shown in Figure 1.



**Figure 1.** Glow curves and time-temperature heating profiles (T) for lithium borate TL detectors after irradiation with 5 mGy  $^{137}\text{Cs}$  gamma rays.

#### *Dosimetric characteristics.*

The summary of dosimetric properties of the thermoluminescence dosimeters investigated is given in Table 1. The results for  $\text{Li}_2\text{B}_4\text{O}_7:\text{Cu,Ag,P}$  are compared to the results for  $\text{Li}_2\text{B}_4\text{O}_7:\text{Cu,In,Ag}$  and  $\text{Li}_2\text{B}_4\text{O}_7:\text{Cu}$  published in our previously paper [6].

TL sensitivities for different detectors, expressed as the number of impulses (corresponding to the glow curve area) per unit dose (impulses/Gy) are given relative to the TL sensitivity of TLD-100. Considerable improvement in the sensitivity was found for the  $\text{Li}_2\text{B}_4\text{O}_7:\text{Cu,Ag,P}$  detector, its sensitivity being about 5 times higher than that of TLD-100.

The reproducibility of calibration factors expressed as  $\pm 1$  standard deviation (SD) in %, is shown in Table 1. The results for  $\text{Li}_2\text{B}_4\text{O}_7:\text{Cu,In,Ag}$  and  $\text{Li}_2\text{B}_4\text{O}_7:\text{Cu,In}$  are from our previous paper [6] and were obtained after irradiation in the dose range 1-15 mGy. Therefore the reproducibility results for these detectors also represent the linearity in this dose range. The results for  $\text{Li}_2\text{B}_4\text{O}_7:\text{Cu,Ag,P}$  were obtained by 10 irradiations of the batch containing 50 dosimeters at a  $^{137}\text{Cs}$  source with the dose of 5 mGy. The linearity for this type of TLD was

determined separately. The linearity, expressed as the standard deviation of calibration factors for samples containing 7 detectors for 6 different doses from 0.1 mGy to 1 Gy was 4%.

Uniformity (expressed as  $\pm 1$  SD in %) in Table 1 represents the variations in sensitivities within the examined batches containing 50 detectors of each type and irradiated to a dose of 2 mGy.

The dose at the lower detection limit,  $D_{LDL}$ , defined as three times the SD of the zero-reading of unirradiated dosimeters [6], was also determined for every dosimeter type.

The protocol and the results of fading for  $Li_2B_4O_7:Cu,In$  and  $Li_2B_4O_7:Cu,In,Ag$  were shown in our previous paper [6].  $Li_2B_4O_7:Cu,Ag,P$  was measured according to the same procedure.

**Table 1.** Comparison of dosimetric characteristics of lithium borate based TL detectors

Dosimeter	$Li_2B_4O_7$ Cu,In	$Li_2B_4O_7$ Cu,In,Ag	$Li_2B_4O_7$ Cu,Ag,P	TLD-100 (LiF:Mg,Ti)
TL sensitivity relative to TLD-100	0.2	1	5	1
Reproducibility of calibration factor	9%	11%	7%	4%
Uniformity	8%	5%	4%	4%
$D_{LDL}$ ( $\mu$ Gy)	43	12	8	5
Fading (14 days)	6%	3%	16%	0%

### *Energy dependence*

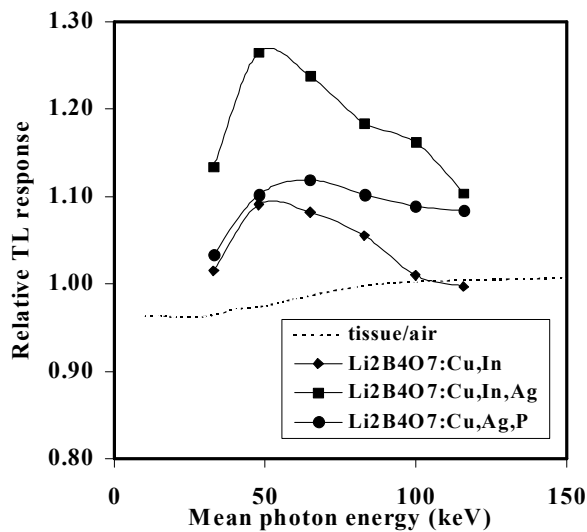
The results for irradiations “free in air” are shown in Figure 2. Measured energy responses relative to air, normalised to  $^{137}Cs$  photons were compared with calculated ratios of mass-energy absorption coefficients for ICRU muscle tissue and air. The results for irradiations on phantom, the relative responses with regard to  $H_p(10, 0^\circ)$  as a function of the mean photon energy, are shown in Figure 3.

### **Conclusions**

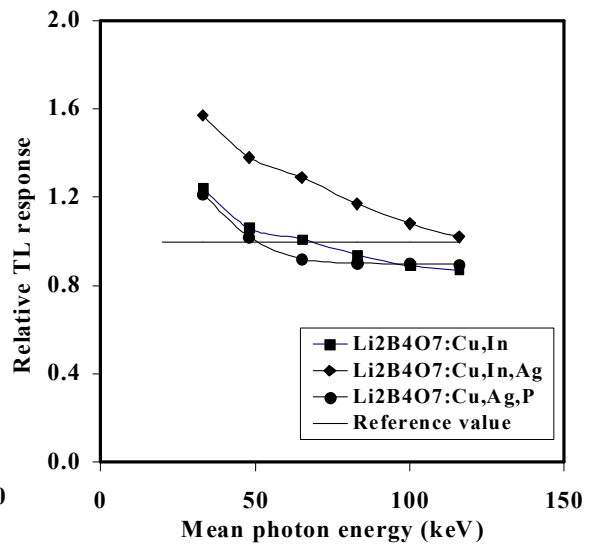
The results for lithium borate based dosimeters show that they are promising new types of tissue-equivalent detectors. The newest of them,  $Li_2B_4O_7:Cu,Ag,P$  shows some improvements in comparison with two earlier types, the most important being the higher sensitivity, even 5 times higher than that of TLD-100. The fading does not meet the requirements of the IEC Standard [5], therefore the improvement of fading correction needs further investigations. At the same time tissue equivalence of this type is also very good: for



irradiations in air it is better than for the other two types and for irradiation on phantom it is better than  $\text{Li}_2\text{B}_4\text{O}_7:\text{Cu,In,Ag}$  and approximately the same as for  $\text{Li}_2\text{B}_4\text{O}_7:\text{Cu,In}$  detector.



**Figure 2.** Measured energy responses of lithium borate TLDs relative to air kerma compared with the calculated ratios of mass-energy absorption coefficients for ICRU muscle tissue and air (normalised to  $^{137}\text{Cs}$  gamma photons).



**Figure 3.** The relative responses of lithium borate TLDs with regard to  $H_p(10, 0^0)$  as a function of the mean photon energy. The response was normalised for  $^{137}\text{Cs}$  gamma radiation.

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# INTERNAL CONTAMINATION WITH $^{241}\text{Am}$ DURING HANDLING OF RADIOACTIVE WASTE.

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## **Introduction.**

A group of workers was internally contaminated with  $^{241}\text{Am}$  during dismantling and treatment procedures when preparing contaminated hermetic boxes for cementation into waste drums. In the past, the hermetic boxes had been used for manufacturing of smoke detectors. The whole period of the use of the boxes had been about 30 years. Americium oxide from Russia was the basic material from which the briquettes were produced which were encapsulated by rolling into gold and silver foil in the process of powder metallurgy and afterwards cut for the use as the radioactive sources in smoke detectors.

The hermetic boxes had not been used any more in the last few years and they became a source of contamination in the laboratory, causing internal contamination of workers too. As to hinder the spread of radioactive material, they were covered by fixating lacquer and by plastic sheets before the dismantling. The whole process of transportation of the boxes from the original place of use to the waste management facility and process of the fragmentation and cementation into drums took three weeks' time in June and July 2001. In the end of this period, a release of  $^{241}\text{Am}$  into working environment was detected. Because significant activity of  $^{241}\text{Am}$  was found on respirators of 3 workers, a suspicion of an internal contamination of a few workers had arisen. It was found, that respirators and gloves were used repeatedly, and between individual operations, they were stored in plastic bags, so that secondary contamination of them during the storage was also possible (1).

## **Measurement of $^{241}\text{Am}$ activity in vivo and in excreta.**

Altogether 19 workers were measured for  $^{241}\text{Am}$  by a whole body counter with HPGe detector and their heads and knees were measured with special LEGe detectors, calibrated for

$^{241}\text{Am}$  in the skeleton. Also, ad hoc measurement, without previous calibration of the thorax with the aim to estimate activity in the lungs, was performed.

The results of the in vivo measurements gave in 7 workers high estimation of the activity in the body with the estimates of committed effective dose of few hundreds mSv. From the very beginning, there was a suspicion that significant activity on the skin all over the body was present, however, not measurable by the detection instruments (Digital Contamination Monitor LB 1210 E) for the surface activity. The surface contamination was proven when  $^{241}\text{Am}$  activity was detected in the sample of cut hair and on the bedclothes of one worker. Another proof of skin contamination was that the activity measured in the heads decreased instead of expected increase according to the ICRP 78 (2) model (fig.1).

Activities of  $^{241}\text{Am}$  excreted with faeces were determined by gamma spectrometry, activities in urine were determined by chemical separation followed by alpha spectrometry.

### **Estimation of intakes and committed effective doses.**

The estimation of the dose was complicated – apart of the skin contamination in short periods after the incident - by the uncertainty of the time of the intakes, by the unknown way of intakes (ingestion vs. inhalation), possible repeated smaller intakes and unknown chemical and physical form of the aerosol. The coefficient for the dose per unit intake  $2.7 \cdot 10^{-2}$  [Sv/Bq] from ICRP 78 (workers, inhalation of class M particles with AMAD 5  $\mu\text{m}$ ) (2) was used for the evaluation. Because of the irremovable contamination of the skin, all realistic estimations of the intakes and of the committed effective doses have to be based on the measurement of  $^{241}\text{Am}$  activity in the excretion only.

The intakes  $I$  (Bq) were calculated according to the formula

$$I = \frac{1}{n} \sum_{i=1}^n \frac{M_i}{m_i(t)} \quad (1)$$

where  $M_i$  is measured activity in daily excretion (i-th sample),  $m_i(t)$  is daily excretion for intake 1 Bq, calculated from model,  $n$  is number of measurements of urine resp. faeces.

The intakes were calculated separately for the excretion with faeces and with urine; they are presented in the table 1.

Time-course of  $^{241}\text{Am}$  activity in the excretion together with the fitted model curves are in figs 2a and b. In 4 cases, increase of  $^{241}\text{Am}$  activity in excreted faeces was observed. It could be caused by new intakes. Time of new intakes were chosen either according to information supplied by radiation protection officer, or, when such data were not supplied, midpoints of monitoring intervals.

Table 1 Estimated intakes of <sup>241</sup>Am and committed effective doses (CED).

Worker		MS	PV	VV	KH	VS	LZ	FH
Faeces	Intake [kBq]	1,04 (1,4) (3,5) (5,1)	2,2	3,2 0,2 (0,4) (2,9)	3,9 (0,9) (0,7)	1,7	1,3	3,1 0,05 0,05 0,08 (13,7) (3,6) (4,0)
	Total Intake [kBq]	1,0	2,2	3,4	3,9	1,7	1,3	3,3
	CED [mSv]	28 - -	60	86 7 - -	106 -	45	34	84 1 1 2 - -
	Total CED [mSv]	28	60	93	106	45	34	88
Urine	Intake [kBq]	1,1	0,2	0,3	0,9	0,2	1,5	2,7
	CED [mSv]	30	7	8	25	5	40	73

### Treatment of contaminated workers with DTPA.

The first estimates of committed affective dose of seven workers were well above 20 mSv what is the value supposed to be appropriate for the administration of DTPA for enhancement of excretion. Therefore, six workers underwent treatment with DTPA administration( two times 1g of Ca DTPA intravenously during 3 days hospitalisation) in the Department for Occupational Medicine of the General Teaching Hospital in Prague. The evaluation of efficiency of the treatment has to be based according to (3) on the fraction of the total incorporated activity which is removed from the body by the treatment, i.e.

$$\frac{(\text{actual} - \text{predicted}) \text{excretion}}{\text{uptake}} \cdot 100 \quad (2)$$

Uptake was not directly measurable. It was supposed that retention in the liver and skeleton are the major part of the uptake. Uptake was calculated from the intake by multiplying it by retention fractions for liver and skeleton. For this purpose the intake was calculated solely from the activity of <sup>241</sup>Am excreted with the urine.

In two cases urine was not collected before the treatment, three cases didn't collect urine after the treatment. For the calculation of the excess of excreted activity the missing values were substituted by the measured values before or after treatment. Results of the calculation of the DTPA treatment are summarized in the table 2.

Table 2 Estimation of the DTPA administration efficiency

Worker	MS	PV	KH	VS	LZ	FH
Before DTPA [Bq/d]	0,085	0,004	0,04	0,050	n/a	n/a
Days of sampling	2	2	3	3	-	-
After DTPA [Bq/d]	0,52	0,055	0,53	0,096	0,89	0,99
Days of sampling	4	4	4	3	3	3
Longer time after DTPA adm [Bq/d]	0,135	n/a	n/a	n/a	0,13	0,48
Days of sampling.	8	-	-	-	3	3
Excess of excreted activity [Bq]	1,6	0,22	1,9	0,14	2,3	3,0
Efficiency according to (2) [%]	6,0	3,4	7,1	1,1	7,4	2,8

## Results and discussion

The workers have been followed for nearly two years after the first intake. Unfortunately, new intakes occurred during this time what further complicated the evaluation of intakes. Estimation of intakes and committed effective doses for periods for which no additional data were supplied, were performed in a conservative way placing the intake in the mid – point of time period since the last measurement. It seems from the time courses of  $^{241}\text{Am}$  excreted in urine that real later intakes are smaller than the estimated ones, therefore this estimated intakes are given in brackets in table 1 and committed effective doses were not calculated from such intakes. In some cases, there is a large discrepancy between calculated intakes from measurements of urine and faeces (table 1) as the significant part of  $^{241}\text{Am}$  was excreted with faeces. This could be caused by the large size of inhaled aerosols, This assumption is supported by results of measurements of aerosols collected with cascade impactor at the workplace half an year after the incident. It was found that more than 87% of aerosols with  $^{241}\text{Am}$  had AMAD higher than  $10.7\ \mu\text{m}$  (4). It is clear that for the presented cases use of standard ICRP 78 model is not good enough. The changes of input parameters (aerosol class and size) as well as change of some parameters within the model – e.g.  $f_1$  (absorption from small intestine to blood), could decrease large discrepancies in the intakes calculated from  $^{241}\text{Am}$  excreted by faeces and by urine. Combination with the biokinetic model for the aerosols of the class S could be helpful too (5).

As to the estimation of the efficiency of DTPA treatment, it is rather complicated because the main part of  $^{241}\text{Am}$  activity was excreted with faeces. It follows from the results in the table 2 that the decrease of the body content and therefore decrease of the health risk with the DTPA administration was relatively small in presented cases. The uncertainty of this estimation is however quite large.

## Conclusions.

Cases of the intakes of radionuclides are quite rare, what has probably lead to the neglect from the side of radiation protection officers and workers themselves too. It is necessary to stress that small intakes of transuranium radionuclides cause already significant committed effective dose and that such intakes are not measurable by in vivo technique. Similar cases appeared time to time in other countries too, in (6) they use a term “alpha awareness“ suggesting that management and operation personnel must be aware of the risk and consequences of the alpha contamination. Generally, decommissioning is the most common process in which alpha contamination occurs not only in the Czech Republic; it has been reported from other countries too. The cause for it is probably the long time period since the operation of facility had ceased which could result in loss of direct knowledge of facility (7) and in addition to it, when commercial company is hired, lack of comprehensive knowledge of the facility.

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Fig.1 Activity of  $^{241}\text{Am}$  in head ( worker MS)

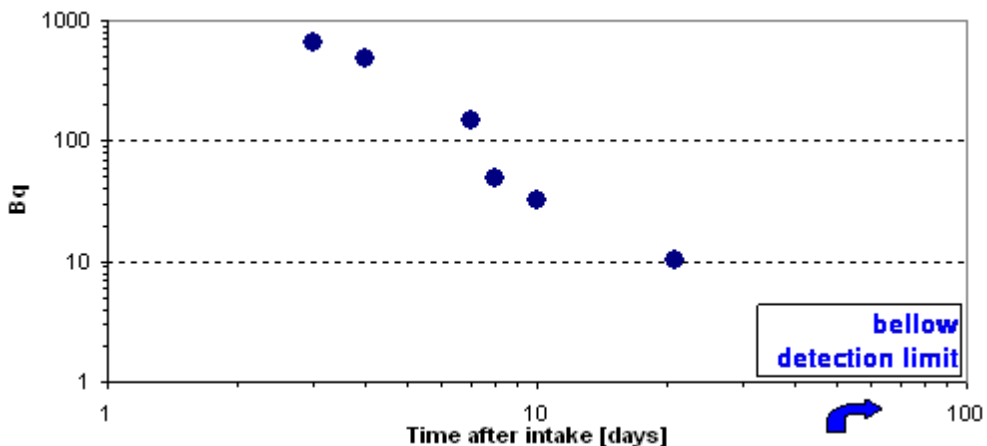


Fig.2a Worker MS, experimental values fitted by model ICRP78.  
From faeces - 4 intakes calculated.  
From urine up to the 2nd intake calculated intake 1690 Bq.

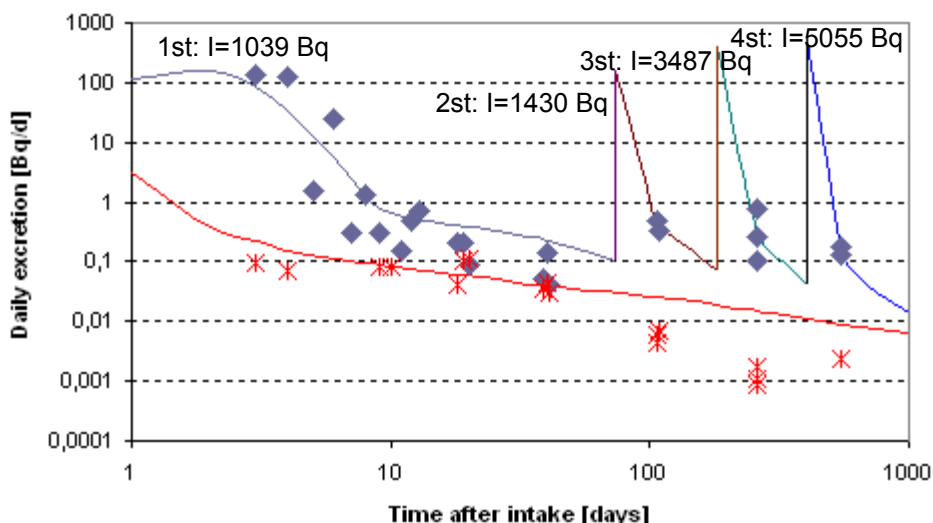
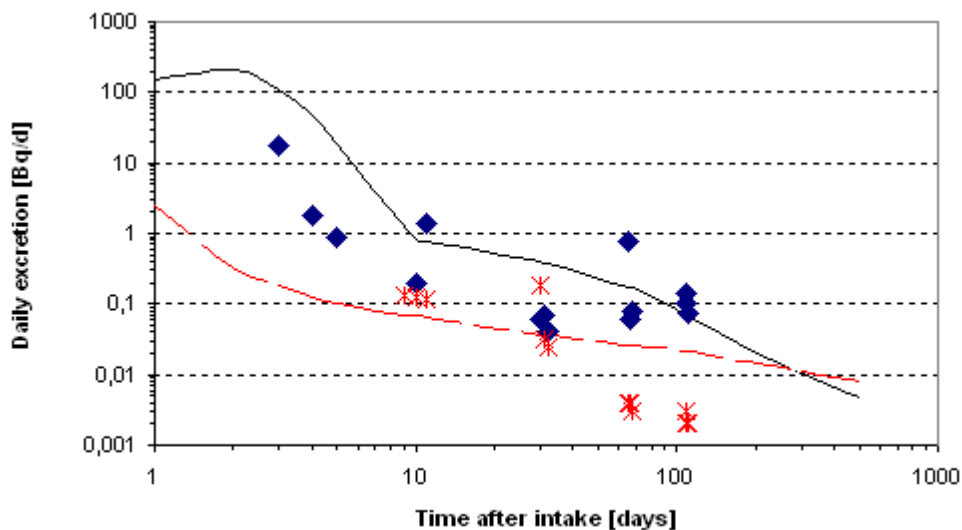


Fig.2b Worker LZ, experimental values fitted by model ICRP78.  
Intake calculated from urine and faeces 1400 Bq.



# WORKERS' DOSES IN CENTRAL EUROPEAN PWR NPPs

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## **Abstract**

As stated in [1] the ISOE database which was established in 1992 forms an excellent basis for studies and comparisons of occupational exposure data between nuclear power plants. In the year 2001 [2] 69% of all participating reactors were pressurised water reactors. The ISOE database presents workers' exposure from 213 participating pressurised reactors from (PWR) 27 countries in that year. Among these 32 PWRs belong to six Central European Countries. The analysis of the exposure of workers based on radiation protection performance indicators (collective dose, average dose etc.) in these PWRs could be related to some nuclear safety performance indicators for recent years using ISOE database. The comparison is made to ISOE world – wide data.

## **Introduction**

In order to perform analysis of radiation protection of workers the analysis of measurements of individual exposure in nuclear power plants is well-established tool in operator practice all over the world. Regulatory authorities regulate the methodology of dosimetry services, which perform individual dosimetry in NPPs, the authorization of services as well as the practice of individual monitoring at NPP sites. In addition, the dose record management is usually treated with special caution due to the fact that the workers can be exposed in different facilities and also in different countries [3,4]. In some NPPs dosimetry services are a part of a plant. Today the following dosimeters or methods are routinely used in NPPs: TLDs, films or electronic dosimeters for assessment of the external exposure of workers and WBC measurements, bioassay measurements for the internal exposure. In order to perform suitable radiation protection of workers the external exposure in NPP is usually measured with passive dosimeters as well as with operational ones. Up to know only few countries authorised the electronic dosimeter as a suitable dosimeter for measurement official external dose.

In order to use an appropriate method for measurement of exposure the radiation sources should be well known. Today around 450 nuclear power plants are operating all over the world, the majority of them with PWRs. As stated in [5] the exposure in PWR NPP is mainly due to the gamma radiation fields while the internal doses are rare. During the maintenance work the exposure is mainly caused by activation products, mainly  $^{58}\text{Co}$  and  $^{60}\text{Co}$ . The exposure from short lived radionuclide  $^{16}\text{N}$ , which is produced by absorption of neutron by  $^{16}\text{O}$ , and exposure from the neutron radiation are important only in cases when workers enter a containment during a reactor operation. In order to make the exposure as low as reasonable achievable operators prepare dose maps of a site as well as the pre and post job analyses of



occupational doses. The list of typical high dose jobs is given for example in [6]. As stated in [6] around 80% of all exposure is usually due to the high dose jobs, which can be estimated to represent only 20% of all jobs in a NPP. Based on the lack of experiences with those critical jobs and also with the dose management, the operators of NPPs and regulatory authorities established in 1992 the ISOE database [6] with the help of the NEA and the IAEA.

The ISOE database presents workers' exposure from 213 participating pressurised reactors from 27 countries in the year 2001. The average annual collective dose in PWRs decreased from 2.01 man Sv in the year 1992 to 0.91 man Sv in the year 2001. The annual collective dose normalised by electrical output for PWRs was in the year 1992 0.32 man Sv/TWh and 0.12 man Sv/TWh in the year 2001. Those data show the influence of the good practice after the development of the ISOE database.

The database is a comprehensive source of data regarding type of reactor, annual collective dose, contract workers doses etc. From the last UNSCEAR data [7] it can be seen that the average annual effective dose of workers in nuclear fuel cycle is still much higher than in other fields where man-made sources are used. The extensive use of the ISOE database by regulatory authorities and operators could lead to the further improvement of radiation protection in NPPs.

### Analysis

In six Central European Countries, Czech Republic, Germany, Hungary, Slovakia, Slovenia and Switzerland, altogether 40 reactor operated in 2001, among them 32 are PWR NPP. The Table 1 shows the number of operating PWR in these countries. In Czech Republic, Hungary, Slovenia and Slovakia no other types of NPP were operating.

Table 1: Number of operating PWRs in Central European Countries.

Country	No. of PWRs
Czech Republic	4
Germany	14
Hungary	4
Slovakia	6
Slovenia	1
Switzerland	3

Radiation protection of NPPs strongly depends on the work management of a plant, technical capabilities of units as well as on the legislation related to these issues in a country. According to the status from April 2002 given in [7], the updated legislation related to the radiation protection of the above-mentioned countries was already implemented in all except in Czech Republic and Slovenia.

As one of important performance indicator of radiation protection unit in NPP the average annual collective dose per operating reactor is shown in Figure 1 for the above-mentioned countries. The figure shows the data from the year 1996 to 2001. The highest value of the data is for the year 2000 for the Krško NPP in Slovenia when both steam generators were replaced.

The replacement contributed 1.48 man Sv in that year. The collective dose of the Krško NPP in the year 2001 was already similar to the values reported by other countries.

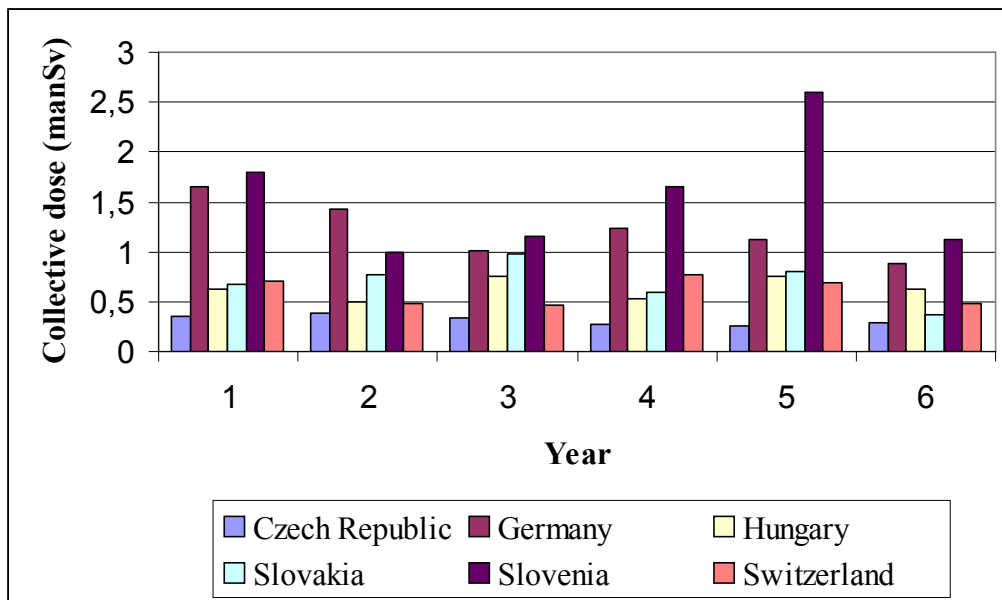


Figure 1. The annual collective doses per operating PWR are given for six Central European Countries.

## Conclusions

In the six Central European Countries altogether 32 PWR operated in the year 2001. The international databases of performance indicators related to radiation protection as for example the ISOE or the UNSCEAR database [8] can be used as an efficient tool in the management of radiation protection of workers in nuclear facilities and regulatory bodies. The databases enable the study of performance trends and the improvement of radiation protection.

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## **IRPA Regional Congress on Radiation Protection in Central Europe**

**Bratislava**

**22.9. – 26.9.2003**

### **THE EXPERIENCE FROM OPERATION OF ELECTRONIC PERSONAL DOSIMETRY SYSTEM AT DUKOVANY, TEMELÍN AND MOCHOVCE NPPs AFTER REPAIR OF SIEMENS DOSEMETERS ELIMINATING FALSE DOSES**

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**RNDr. Milan Zrubec, Ing. Henrich Kaiser - SE, a.s. – Mochovce NPP**





### HISTORY OF SEOD SYSTEM

- 1999 SEOD Development
- 2000 Dukovany NPP MGPI
- 2001 Temelín NPP Siemens
- 2001 Mochovce NPP Siemens
- 2003 VYZ Jaslovské Bohunice NPP MGPI

This presentation summarizes the operational experience of the Electronic Personal Dosimetry Systems installed at Dukovany, Temelín and Mochovce NPPs.



## DESCRIPTION OF SYSTEM

The system consists of three basic parts:

- Electronic personal dosimeters (EPD)
- Physical layer (HW)
- Logical layer (SW)

Pobyty v KP					
Vstup	Místo	Číslo FD	Příjmení	Objekt	
04.06.2002 22:45:58	CDRK	53745	Chvátal	5	
04.06.2002 18:40:43	CDRK	K7941	Coufal	2	
04.06.2002 18:39:25	CDRK	K8203	Ferda	2	
04.06.2002 22:28:47	DRK2	K8547	Sedláček	2	
04.06.2002 22:29:14	DRK1	K8120	Němec	2	
04.06.2002 22:43:25	CDRK	K6306	Horák	2	
04.06.2002 22:43:34	CDRK	K6335	Pišťák	2	
04.06.2002 22:44:53	CDRK	K6329	Oplatek	2	
04.06.2002 22:43:12	CDRK	K6310	Chmelář	2	



## ELECTRONIC PERSONAL DOSEMETERS (EDP)

Parameter	DMC 90	DMC 2000	Siemens Mk.1	Siemens Mk.2
Measuring range Hp(10)	1 $\mu$ Sv - 10 Sv	1 $\mu$ Sv - 10 Sv	1 $\mu$ Sv - 16 Sv	1 $\mu$ Sv - 16 Sv
Measuring range Hp'(10)	10 $\mu$ Sv/h - 10 Sv/h	0,1 $\mu$ Sv/h - 10 Sv/h	10 $\mu$ Sv/h - 16 Sv/h	10 $\mu$ Sv/h - 4 Sv/h
Energy range	60 keV - 3 MeV	$\gamma$ : 20 keV - 6 MeV $\beta$ : 60 keV - 2,3 MeV	$\gamma$ : 20 keV - 10 MeV $\beta$ : 250 keV - 1,5 MeV	$\gamma$ : 15 keV - 10 MeV $\beta$ : 250 keV - 1,5 MeV





## PHYSICAL LAYER (HW)

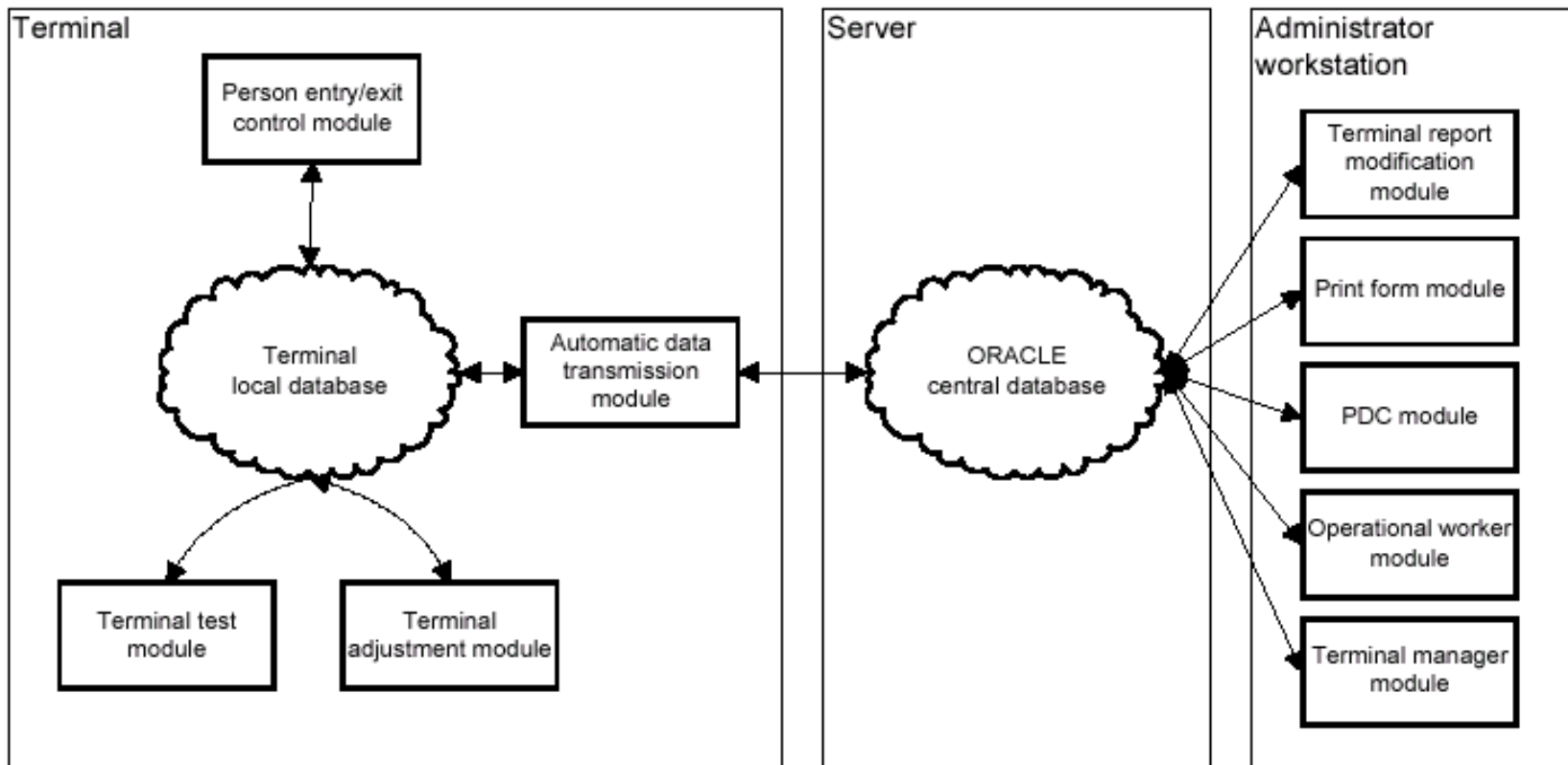
### SEOD TERMINAL (Dukovany NPP)







## LOGICAL LAYER (SW)





## RESISTANCE AGAINST ELECTROMAGNETIC INTERFERENCE

The problem during use of electronic dosimeters is reading false doses caused by electromagnetic fields, e.g. by mobile phones, grinders, welding machines, etc. At Dukovany NPP, the workers which dosimeters could be exposed to electromagnetic interference are provided with additional RPL dosimeters.

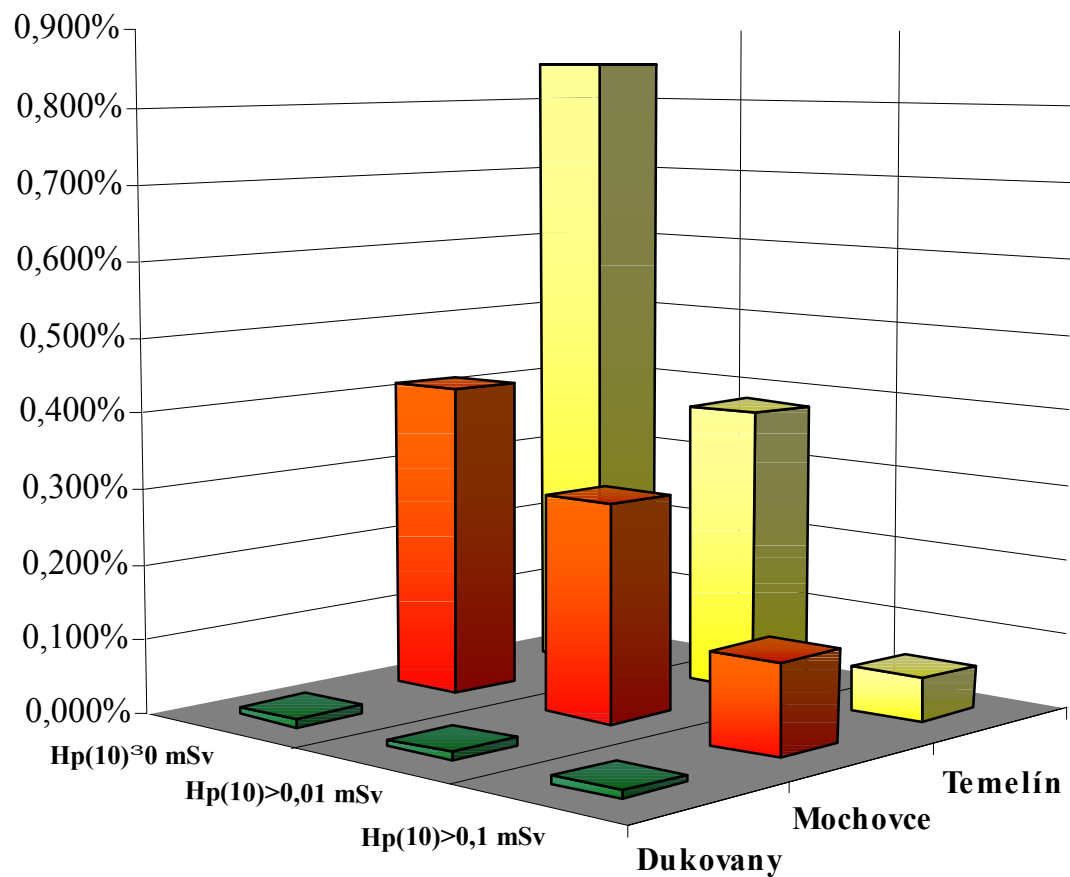
The SEOD system allows searching out and correcting such false doses. Based on these corrections we have performed the analysis of sensitivity for the individual dosimeter types against interference.

The period being compared: from 1/1/2002 to 30/6/2002

	Number of entries	Number of false doses					
		Hp(10) ≥ 0 mSv	%	Hp(10) > 0.01 mSv	%	Hp(10) > 0.1 mSv	%
Dukovany	75 481	8	0.011%	8	0.011%	8	0.011%
Temelín	60 057	510	0.849%	230	0.383%	35	0.058%
Mochovce	63 610	265	0.417%	184	0.289%	77	0.121%
<b>MGP</b>	75 481	8	0.011%	8	0.011%	8	0.011%
<b>Siemens</b>	123 667	775	0.627%	414	0.335%	112	0.091%

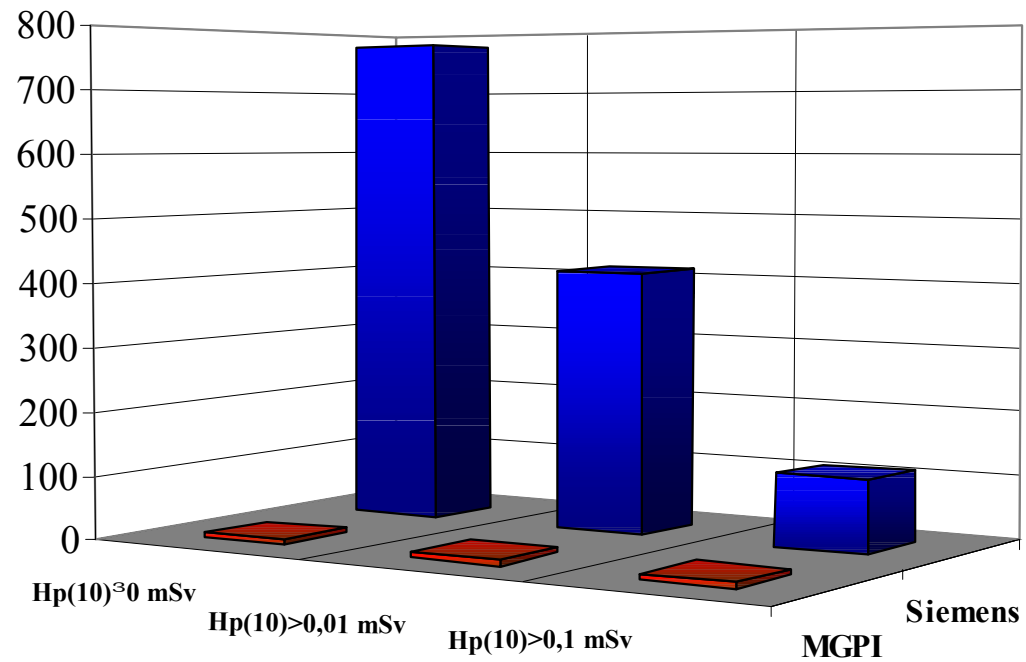


## NUMBER OF FALSE DOSES





## NUMBER OF FALSE DOSES (MGP vs. SIEMENS)





## NUMBER OF FALSE DOSES AFTER CORRECTION

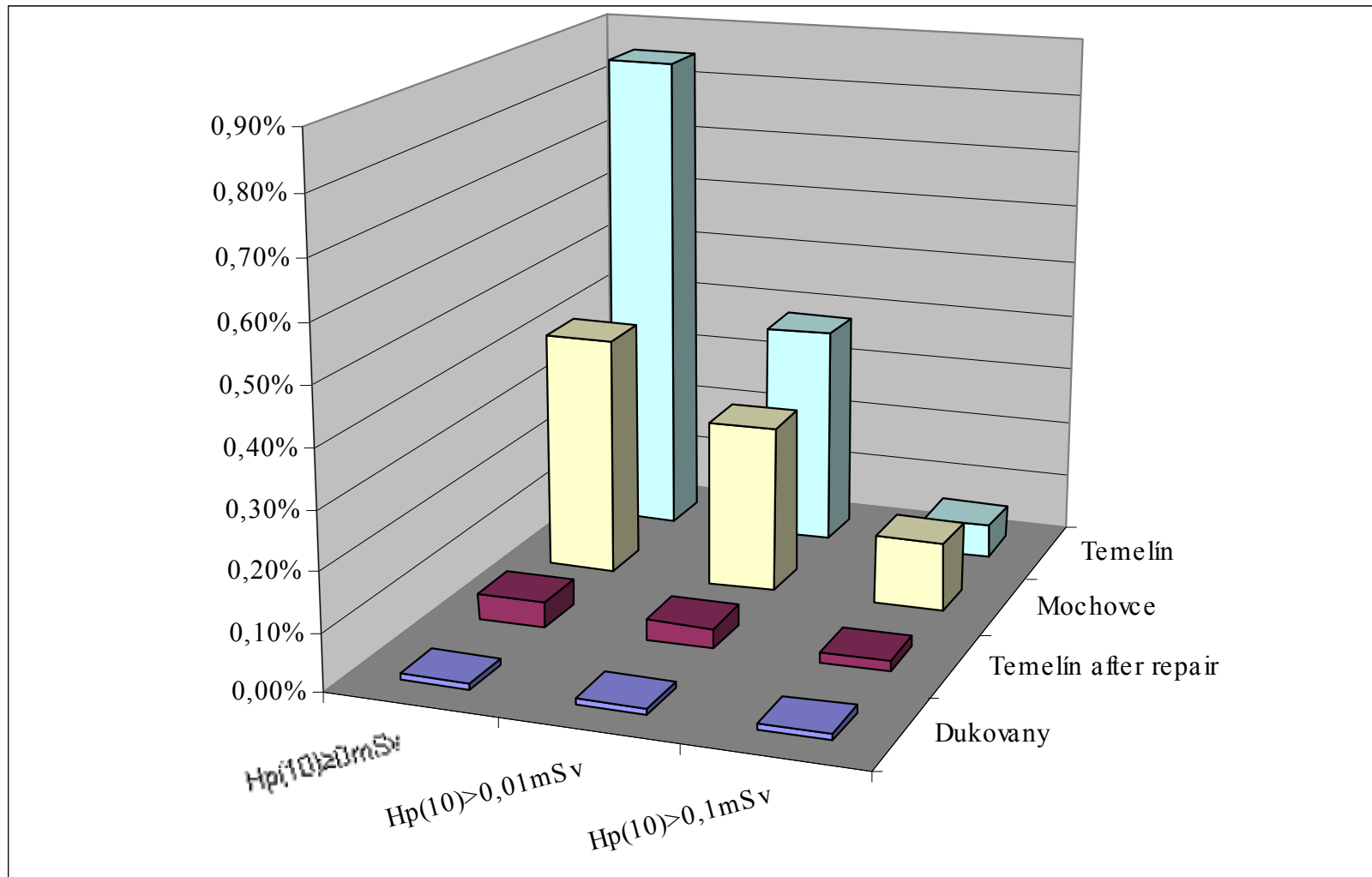
Analysis of manually corrected false doses at Temelín NPP from 1/1/2003 to 30/6/2003 and comparison with the results before the correction.

	Number of entries	Number of false doses					
		Hp(10) $\geq$ 0 mSv	%	Hp(10) > 0.01 mSv	%	Hp(10) > 0.1 mSv	%
Temelín before correction	60 057	510	0.849%	230	0.383%	35	0.058%
Temelín after correction	93 188	41	0.043%	33	0.035%	17	0.018%

False dose reading after correction of Siemens dosimeters is remarkably lower and false dose readings have no significant effect on the total collective dose equivalent.



## NUMBER OF FALSE DOSES AFTER CORRECTION





®

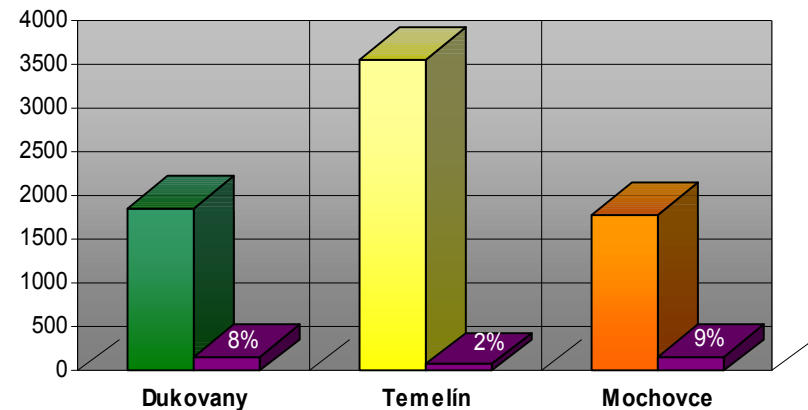
## ELECTRONIC DOSEMETERS VRS. FILM DOSEMETERS

Comparison of electronic dosimetry measurements vrs. film dosimetry measurements.

In the next table, there is one possible presentation of this comparison. We have compared individual personal doses received by film dosimetry versus the doses from electronic dosimetry. Only such doses were considered consistent for which the variation was not higher than 30 % and also doses from the electronic dosimetry which were below the minimum detectable dose of film dosimeter.

Comparison period: 1/1/2002 to 30/6/2002

	Number of monitored persons	Number of persons for which the doses are not consistent	%
Dukovany	1859	156	8 %
Temelín	3567	63	2 %
Mochovce	1790	125	7%





We performed a detailed comparison of the doses received by film dosimetry versus electronic dosimetry at Dukovany NPP over the period from April 2000 to September 2002. Within this period 33 767 film doseimeters was evaluated.

In 2001, we also carried out a control test of film and electronic doseimeters at Temelín NPP, where the doseimeters were exposed to ionizing radiation standards (Cs-137) with doses in a range from 0.1 to 5 mSv.

The variance of real and measured doses for electronic doseimeters has not exceeded 5%, which is better result than for film doseimeters.





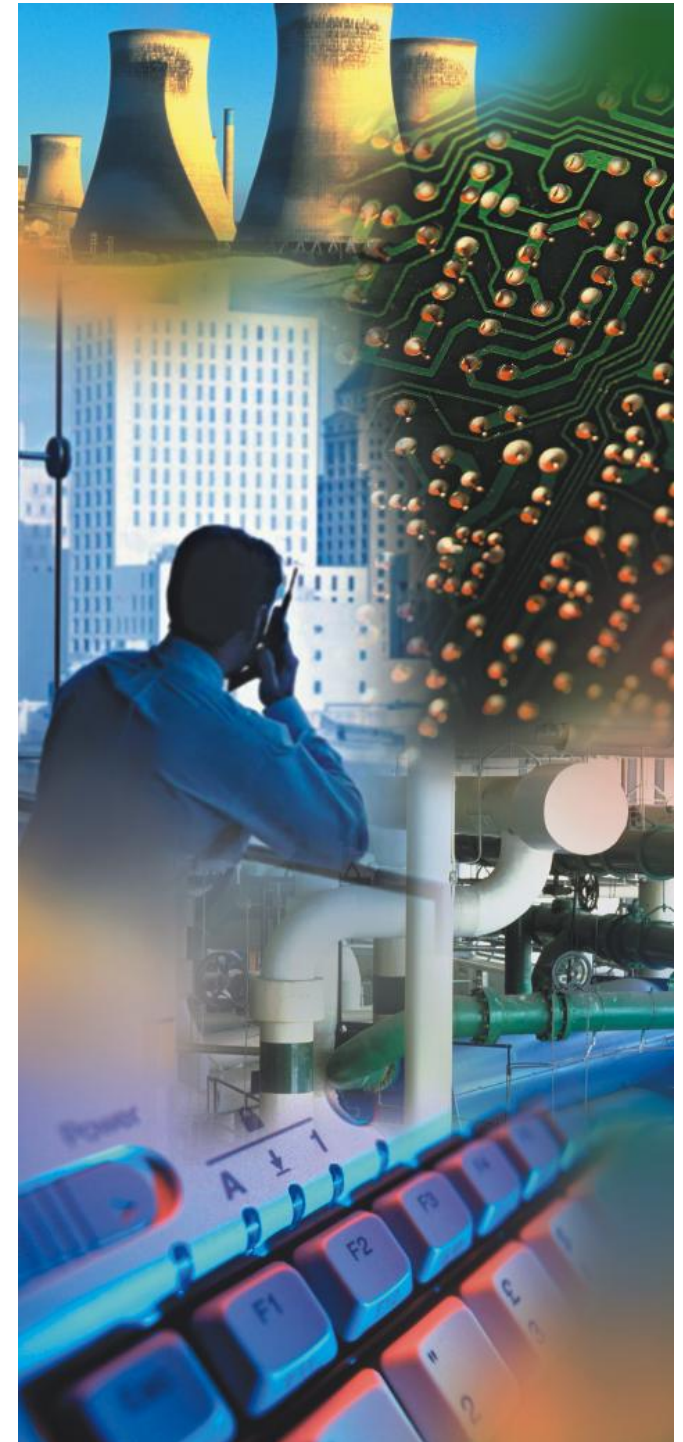
## CONCLUSION

### **In conclusion, we can say:**

1. This presentation has not set a target to carry out a precise comparison of electronic and film dosimetry, but it has demonstrated the possibilities of SEOD system and the possibility of easy dose comparison between the individual NPPs after introducing this electronic dosimetry system.
2. Basically, the results of film and electronic dosimetry systems are according to our findings nearly identical.
3. Electronic dosimeter sensitivity to interfering electromagnetic fields is a problem which is easily re-movable. In addition, if we know this problem, these false doses in the SEOD system can be easily revealed (e.g. by investigation of histograms) and repaired.



**THANK YOU FOR YOUR ATTENTION!**



# VERIFYING THE CAPABILITY OF A RECOMBINATION CHAMBER IN A MIXED FIELD

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## Abstract

Dosimetry in stray radiation fields outside the shielding of hadron accelerators is a difficult task due to the complex nature of the radiation field. An instrument desirable in operational health physics should measure dose equivalent of any composition of radiation components in the entire energy range. A promising candidate for single detector dosimetry in complex radiation fields is a recombination chamber. Performance of a REM-2 recombination chamber was investigated at the CERN-EU high-energy Reference Field (CERF) facility of CERN. The sum of dose equivalents measured with detectors sensitive to only one component of the mixed radiation field was compared with the total dose equivalent obtained with the recombination chamber. In addition the results were compared with Monte Carlo (MC) simulations.

## 1 Introduction

Stray radiation fields outside the shielding of hadron accelerators are of complex nature. They consist of a multiplicity of radiation components (neutrons, photons, electrons, pions, muons, etc.), extending over a wide range of energies. All these particles contribute to the total dose equivalent,  $H$ . This makes dosimetry difficult, since it would be desirable to have one radiation monitor, sensitive to all particles in the entire energy range, which determines correctly the total dose equivalent. An instrument that fulfils these requirements well is the recombination chamber [1].

The performance of a recombination chamber was investigated at the CERN-EU high-energy Reference Field (CERF) facility. CERF provides a mixed field of mainly neutrons and photons ranging from thermal energies up to hundreds of GeV [2]. The energy spectra of the

different particle components are estimated with Monte Carlo (MC) simulations and have been experimentally verified in the past with a range of different instruments [2].

In this work we measured the total dose equivalent at the reference exposure position CT6 of CERF with the recombination chamber and compared it to calculated values. In addition, we measured separately the dose equivalent for low and high LET radiation, with dedicated detectors.

## **2 The CERN-EU High Energy Reference Field (CERF) Facility**

CERF was designed to resemble the radiation environment at subsonic aviation altitudes. It provides a well-known mixed high-energy radiation field for the calibration of various instruments. The reference field is created by a positively charged mixed hadron beam (61% pions, 35% protons, 4% kaons) with a momentum of 120 GeV/c fired onto a copper target, which is located either below 80 cm of concrete or below a 40 cm iron shield [2]. The beam intensity is monitored by a Precision Ionisation Chamber (PIC). One PIC-count corresponds to  $2.2 \times 10^4 \pm 10\%$  primary particles. The secondary particles produced in the copper target traverse the shielding where on top several reference exposure positions are available. At these reference positions, the particle fluences and spectra are well known from simulations with the MC program FLUKA [3,4] and measurements [2].

## **3 Instruments**

For measuring the different components of the total dose equivalent, several instruments were used. The neutron ambient dose equivalent was determined with an extended-range Bonner sphere spectrometer [5,6], with the extended-range REM counter LINUS [7] and with a HANDI TEPC [8]. For measuring the low LET component of the dose equivalent the HANDI TEPC was taken. The REM-2 chamber [1] is capable to determine the total dose equivalent.

### *2.1 Recombination Chamber*

The REM-2 chamber is a high-pressure ionisation chamber filled with a mixture of methane and nitrogen (5%) under a pressure of about 1 MPa. The chamber is operated under the conditions of initial recombination where volume recombination is negligible. The degree of initial recombination can be related to the LET of the radiation. This fact is used to determine the average quality factor of a radiation field. Employing a method developed by [9] and [1] the ratio of the current measured in recombination and saturation mode at

appropriate chosen voltages provides a measure of initial recombination. Combining the quality factor and the measured absorbed dose a total dose equivalent  $H_{tot}$  can be evaluated.

## 2.2 *The LINUS*

The spherical LINUS (Long Interval NeUtron Survey-meter) is an extended-range REM counter consisting of a spherical  $^3\text{He}$  proportional counter within a moderator sphere made of shells of polyethylene, lead and boron doped plastic (details of the construction can be found in [7]). The response function of the LINUS follows very well the fluence to ambient-dose-equivalent curve, especially in the region above 10 MeV, where conventional rem counters underestimate the ambient dose equivalent dramatically. Thus, it can measure directly  $H^*(10)$ .

## 2.3 *Extended-Range Bonner Sphere Spectrometer*

At CERN a new extended-range Bonner sphere spectrometer with sensitivity up to 1 GeV was developed recently [5,6]. It uses a Centronics  $^3\text{He}$  proportional counter connected via a preamplifier to a compact acquisition system from Münchner Apparate Bau – MAB, which can be set and read out by a PC. The counter is coupled, step-by-step, with a set of 7 moderator spheres. 5 moderators are conventionally made of polyethylene. Their diameters are 81 mm, 108 mm, 133 mm, 178 mm and 233 mm, from which the smallest one was also exposed with 1 mm cadmium cover. An overall measuring range of the system with these moderators is limited to ca. 10 MeV. The other two moderators, named Ollio and Stanlio, expand the measuring range up to ca. 1 GeV. 'Ollio' is a sphere with a diameter of 255 mm and consists of shells of 3 cm polyethylene, 1 mm cadmium, 1 cm lead, and 7 cm polyethylene thickness (from the  $^3\text{He}$  proportional counter outwards). 'Stanlio', with a diameter of 118.5 mm, consists of shells of 2 cm polyethylene, 1 mm cadmium, and 2 cm lead thickness. The response functions are given elsewhere [5,6]. The ambient dose equivalent is determined by folding the measured spectrum with fluence to ambient-dose-equivalent conversion coefficients of ICRP74 (up to 19 MeV) [10] and above 19 MeV of Pelliccioni [11].

## 2.4 *HANDI TEPC*

The HANDI-system developed by the Universität des Saarlandes in Homburg is based on a low-pressure tissue equivalent proportional counter (TEPC) [8]. The spherical TEPC is filled with propane based tissue-equivalent gas simulating a density of  $1 \text{ g/cm}^3$  and a diameter

of about 2  $\mu\text{m}$ . The TEPC is operated together with a data processing system based on pulse-height analysis using only 16 channels of nearly logarithmically increasing width. The system is able to measure in real time absorbed dose and the lineal energy  $y$ .

#### 4 Results and Discussion

Measurements of the dose equivalent were carried out at position CT6 of CERF. Emphasis was put on measuring low LET particles (photons and muons) and neutrons. The fluence of other hadrons as well as of electrons and muons, which are originating from beam interactions in the copper target, is much lower than that of neutrons and photons [2]. However, a significant muon component is present, which appears as intensity independent background. This is mainly due to pion decay in the beam line or neighbouring beam lines.

The measurements with the HANDI TEPC and the REM-2 chamber were performed at beam intensities of 2200, 4000, 6000 and 7500 PIC-counts per pulse extracted from the accelerator. Values of the low LET component of the total dose equivalent  $H_{low}$  obtained by the HANDI TEPC are given for each beam intensity in order to see the influence of the muon background. The results of the high LET component of the total dose equivalent  $H_{high}$  obtained by the HANDI were averaged over all beam intensities, as well as  $H_{tot}$  obtained by the REM-2 chamber. Measurements with the LINUS and the Bonner spheres were carried out at one intensity,  $\sim 1500$  PIC-counts per spill. The given uncertainties include statistical errors of the measurements and simulations and calibration uncertainties of the instruments.

Table 1 shows the dose equivalent for low LET radiation normalized to incident beam particle. The photon component was calculated by folding the simulated photon spectrum obtained by FLUKA [2] with the conversion coefficients of [11] for ambient dose and maximum dose equivalent. The background contribution due to the muons was determined by [12] and the entire low LET component by the HANDI TEPC. We see that the sum of  $H_{\mu}$  and  $H_{\gamma}$  reproduces very well the value of the dose equivalent  $H_{low}$  for the entire low LET radiation. Table 1 also points out the consequences of the muon background, which is obviously more important for low intensities. It appears as a non-constancy of the normalized detector responses for different intensities of the incident hadron beam.

Table 2 shows a comparison of the neutron dose equivalent at CT6, measured by the HANDI TEPC, the LINUS and the extended-range Bonner sphere spectrometer and calculated by folding the neutron spectra obtained by FLUKA with the fluence to ambient-dose-equivalent conversion coefficients of ICRP74 (up to 19 MeV) [10] and above 19 MeV of Pelliccioni [11].

The agreement between the different methods is very good. We also see that the sum of  $H_{low}$  and  $H_{high}$ , which corresponds to  $H_{tot}$ , is very well given within the uncertainties directly by the REM-2 chamber.

**Table 1:** Measured low LET and muon components of the dose equivalent and calculated photon component at the position CT6 of CERF for different beam intensities.

<b>Beam Intensity</b> [PIC/pulse]	<b>H<sub>low</sub></b> [pSv/PIC]	<b>H<sub>μ</sub></b> [pSv/PIC]	<b>H<sub>γ</sub></b> [pSv/PIC]
7500	53.9 ± 3.8	24 ± 1	
6000	63.1 ± 6.3	30 ± 2	
4000	78.4 ± 5.5	45 ± 2	
2200	116 ± 12	82 ± 5	
<b>Average</b>	<b>78 ± 33</b>	<b>45 ± 29</b>	
<b>Simulation</b> Amb. Dose Equivalent			<b>17 ± 2</b>
Max. Dose Equivalent			<b>27 ± 3</b>

**Table 2:** Dose equivalent values measured with different instruments and calculated by MC simulations for different components of at position CT6 of CERF.

	<b>H<sub>low</sub></b> [pSv/PIC]	<b>H<sub>high</sub></b> [pSv/PIC]			<b>H<sub>tot</sub></b> [pSv/PIC]
Measurement	HANDI 78 ± 33	HANDI 242 ± 11	LINUS 259 ± 26	Bonner 287 ± 20	REM-2 314 ± 14
Simulation		282 ± 28			

## 5 Conclusions

The recombination chamber represents a radiation monitor sensitive to all particles in a mixed radiation field. The total dose equivalent measured with the recombination chamber agrees very well with the total dose equivalent either obtained by Monte Carlo simulations or coming from the sum of measurements with particular detectors for neutron and low LET radiation. The measurements showed that the recombination chamber alone is able to replace a number of radiation detectors for determining the total dose equivalent.

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# MEASUREMENT AND SIMULATION OF THE IN-FLIGHT RADIATION EXPOSURE ON DIFFERENT AIR ROUTES

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## **Introduction**

With the steadily increasing human mobility and the development of improved high-altitude jet aircraft, the problem of civil aircrew exposure to an elevated level of cosmic radiation becomes a key issue in the field of radiation protection and radiobiology. The accurate assessment of the biologically relevant dose provides the basis for the estimation of radiation risk factors. The experimental assessment of the radiation exposure in the complexly mixed radiation environment encountered at aviation altitudes is usually associated with extensive metrological expenditure in order to cover the entire particle spectrum and energy range. The exposure of aircrew personnel to cosmic radiation is considered to be occupational exposure and requirements for dose assessment are given in the European Council Directive 96/29/Euratom. With the introduction of two categories of occupationally exposed workers it is important to check if the limit of 6 mSv for category B personnel is exceeded.

## **Instrumentation and Methodology**

Thermoluminescent dosimeters (TLDs) are a perfect survey instrument for in-flight application. As passive detectors they do not require air-worthiness certification, as they need no power supply, contain no flammable gases and emit no electromagnetic radiation – and therefore cannot interfere with aircraft electronics. Their small size on the order of some mm<sup>3</sup> and low mass makes them ideal tools for routine usage. In standard dosimetry, TLDs are applied to measure the absorbed dose, but the development of sophisticated experimental techniques – the High-Temperature Ratio and the Pair Method – enables the assessment of the dose-average LET and from it the biologically relevant dose equivalent.

## High-Temperature Ratio Method

The High-Temperature Ratio (HTR) Method for LiF: Mg, Ti TLDs utilizes the well-investigated relative intensity of the combined high-temperature glow peaks 6 and 7 compared with the dominant peak 5 (left-hand side of Figure 1) as an indication of the dose-average LET of a mixed radiation field of unknown composition [1]. Extensive irradiation campaigns with high-energy ions of different  $Z$  ranging from hydrogen to iron established an HTR vs. LET calibration curve (right-hand side of Figure 1) [2]. The HTR Method has been applied previously with great success on several space missions (including measurements onboard Space Station Mir [1-3], space shuttles [4], bio-satellites [5] and the International Space Station [6]) as well as in radiotherapeutic dosimetry [7].

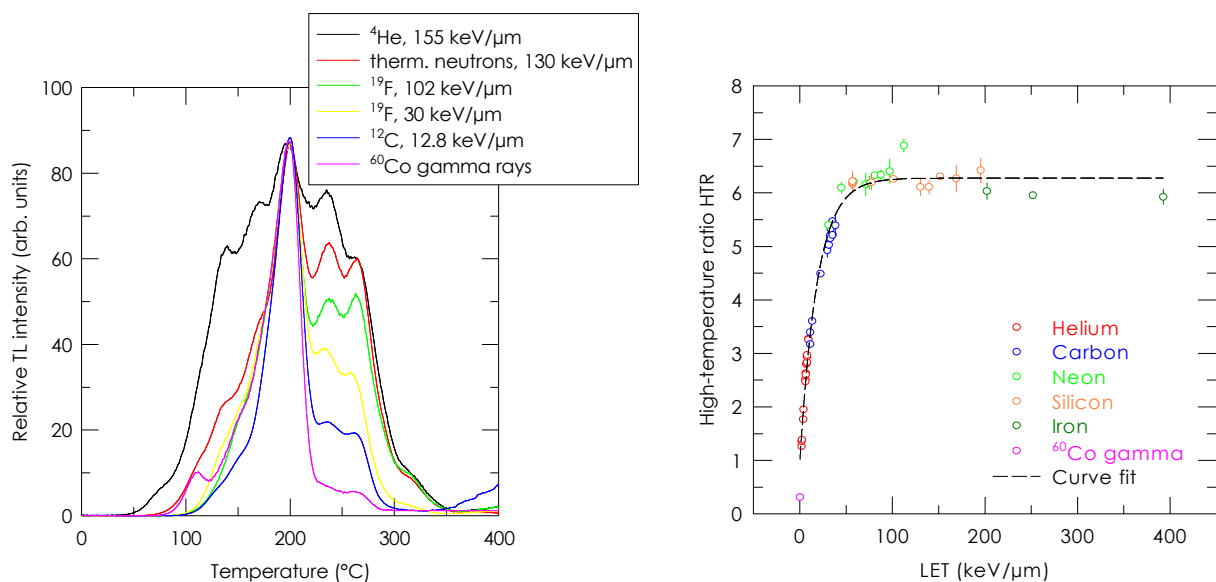


Figure 1. Dependence of high-temperature thermoluminescence (peaks 6, 7) on LET (left); HTR vs. LET calibration curve,  $^{60}\text{Co}$  is shown for reference (right).

## Extended Pair Method

The difference in the peak-5 readings of the neutron-sensitive TLD-600 ( $^6\text{LiF}$ : Mg, Ti) and the neutron-insensitive TLD-700 ( $^7\text{LiF}$ : Mg, Ti) can be utilized to assess the neutron dose equivalent accumulated in-flight. For this purpose, the dosimeter crystals were calibrated individually in the CERN-EU High Energy Reference Field (CERF) [8] which simulates the cosmic-ray induced neutron spectrum in good detail.

## Results and Discussion

The experiments conducted onboard passenger aircraft on different north-bound and trans-equatorial flight routes were aimed at the following: to measure the total dose equivalent

accumulated during the flight, to assess the contribution of neutrons, and to compare the results with calculations by means of the well-known CARI computer code. Measurements were performed on a series of eight north-bound flights between Cologne and Washington as well as on the routes Vienna-Atlanta, Vienna-Sydney and Vienna-Tokyo during different solar activity conditions. Precise altitude and route profiles were recorded by the pilots. For the Vienna-Sydney return flight, the flight log data file could be used by courtesy of Lauda Air.

Table 1. Total dose equivalent rate, neutron dose equivalent rate and the CARI-6M-calculated total dose equivalent rate for different flight routes.

Flight route and date	$\dot{H}_{\text{total}}$ ( $\mu\text{Sv/h}$ )	$\dot{H}_{\text{neutron}}$ ( $\mu\text{Sv/h}$ )	Neutron contribution (%)	$\dot{H}_{\text{CARI}}$ ( $\mu\text{Sv/h}$ )
CGN-IAD Jun./Jul. 1996	$6.7 \pm 0.4$	$3.9 \pm 0.8$	58.2	5.9
VIE-ATL Nov. 2001	$4.1 \pm 0.4$	$1.7 \pm 0.2$	41.4	3.9
VIE-SYD Oct./Nov. 2001	$2.1 \pm 0.1$	$0.7 \pm 0.1$	33.3	2.4
VIE-NRT Jan. 2003	$4.0 \pm 0.3$	$2.7 \pm 0.3$	67.5	4.2

The results of the dosimetric investigations are summarized in Table 1, containing the total dose equivalent rate, the contribution of the neutron component and the CARI-6M-calculated dose for each air route. The highest dose equivalent rate of  $6.7 \pm 0.4 \mu\text{Sv/h}$  with a 58.2 % contribution of neutrons was found for the north-bound flight series Cologne-Washington during the solar minimum conditions of June/July 1996. At minimum solar activity, the Earth is less shielded from cosmic-ray particles by the interplanetary magnetic field. The dose rate value of  $6.7 \mu\text{Sv/h}$  achieved for this trans-atlantic flight route may therefore be regarded as an upper limit for the radiation exposure of pilots and cabin crew. Measurements onboard a comparable return flight between Vienna and Atlanta around the maximum of the 23<sup>rd</sup> solar activity cycle revealed a dose rate of  $4.1 \pm 0.4 \mu\text{Sv/h}$  and thereby confirmed the solar influence. The most pronounced contribution of neutrons in the dose equivalent (67.5 %) was found for the route Vienna-Tokyo which was operated at very high latitudes of up to 69 °N. The latitude-dependence of the radiation exposure was demonstrated by means of experiments onboard a return flight between Vienna and Sydney in November 2001. The assessed dose equivalent rate was  $2.1 \pm 0.1 \mu\text{Sv/h}$  with an only 33.3 % contribution

of neutrons. Due to the shielding effect of the Earth's magnetic field, only a relatively small part of cosmic radiation can penetrate the atmosphere in the equatorial region.

The experimental results were compared with model calculations using the latest release 6M of the CARI code. Precise altitude and route data on a ten-minute to one-hour scale were taken as input. The calculated dose values indicate that the algorithms employed for the computational assessment of route doses have been significantly improved during the last decade. The CARI results generally tend to be in reasonable agreement with the measured values, although the doses for north-bound flights during solar minimum might still be underestimated by up to 15 %. The most important insufficiency in all computational approaches regards the effects of major solar particle events (SPEs) presenting a serious danger primarily for future high-altitude and polar-orbital flights in causing severe biological hazards. The frequency of these irregular events corresponds to the solar activity cycle. This fact is taken into account in the codes by semi-empirical models which certainly have to fail in forecasting accurate dose values for a specific flight. Therefore, dosimetric surveillance of aircrew members would be essential and cannot be completely replaced by calculations.

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# SPACE RADIATION DOSIMETRY ONBOARD MIR AND ISS OVERVIEW AND COMPARISON \*

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## **Introduction**

The Atomic Institute of the Austrian Universities looks back on a successful 12-year tradition in space dosimetry, conducted mainly in co-operation with the Institute of Biomedical Problems (IBMP), Moscow/Russia. The radiation environment in space consists mainly of protons, electrons and heavy charged particles (HCPs) and interacts with the hull of a spacecraft, thereby producing a complex secondary radiation field that additionally includes neutrons, photons, muons and pions. The major issue of our studies concerns the development of highly precise instrumentation to provide a basis for a reliable estimation of hazardous radiation effects in astro- and cosmonauts. The paper gives an overview of the related missions with emphasis laid on the phantom experiment onboard Mir station and the first Austrian dose measurements onboard the International Space Station (ISS).

## **Instruments and Methods**

Radiation doses were determined by means of thermoluminescent dosimeters (TLDs) of different types which have been calibrated individually prior to and after the flights. Read-out was processed with the laboratory-made TL-DAT.II system facilitating adjustable heating rates up to 480 °C. In routine dosimetry, TLDs are used to measure the absorbed dose, but the development of sophisticated experimental techniques enables the assessment of the dose-average LET and from it the biologically relevant dose equivalent. By application of different TLD types, information on the neutron dose can be achieved.

## **High-Temperature Ratio Method**

The High-Temperature Ratio (HTR) Method for LiF: Mg, Ti TLDs uses the well-investigated relative intensity of the combined high-temperature glow peaks 6 and 7

compared with the dominant peak 5 as an indication of the dose-average LET of a mixed radiation field of unknown composition [1]. Extensive irradiations in various ion beams ranging from hydrogen to iron established a HTR vs. LET calibration curve [2]. The HTR Method has been applied with great success for the determination of the biologically relevant dose in mixed radiation environments during more than 3000 days of measurements in space onboard Mir [1-3], space shuttles [4], bio-satellites [5] and the ISS. In addition, the method was used for LET determination onboard aircraft [6-7] and in radiotherapeutic dosimetry [8].

### **Extended Pair Method**

The difference in the peak-5 readings of the neutron-sensitive TLD-600 ( $^6\text{LiF}$ : Mg, Ti) and the neutron-insensitive TLD-700 ( $^7\text{LiF}$ : Mg, Ti) can be utilized to determine the dose from thermal and epithermal neutrons (Pair Method). Based on a calibration in the CERN-EU High-Energy Reference Field (CERF) [9] which closely resembles the neutron spectrum at high altitudes and in space, the Pair Method was extended to assess the dose from neutrons with energies up to several GeV.

### **Project “Phantom” Onboard Space Station Mir**

The project “Phantom” investigated for the first time the depth dose profile over a long-term stay onboard a space station. The exposure times during the three project phases ranged from 99 to 271 days, starting in May 1997. The spherical water-filled phantom with a diameter of 35 cm was developed by the IBMP. It facilitates four channels for dosimeter insertion in the same plane and orientated perpendicular to each other. Due to their small size of  $6.4 \times 6.4 \times 0.89 \text{ mm}^3$  TLDs are perfect devices for these studies. The phantom body was positioned in the Commander and Board Engineer Cabins (project phases 1 and 2) as well as in the Module Kvant-2 (phase 3). Therefore, the depth dose distribution could be investigated under different shielding conditions.

Figure 1 displays the summarized results for the depth distribution of absorbed dose and dose equivalent. A stronger gradient of absorbed dose is observed for the plane perpendicular to the spacecraft hull (project phases 1 and 2). The absorbed dose rates are almost independent of the phantom location inside the station (Commander and Board Engineer Cabin). During the project phase 3 the phantom was installed in the Kvant-2 Module and the TLDs were inserted into the channels orientated parallel to the spacecraft hull. Due to the increased shielding thickness of some additional  $5 \text{ g/cm}^2$  in the Kvant-2 Module the observed absorbed dose rates are generally lower. Using the Pair Method it could be observed

that the heavier shielding produces additional neutrons. This trend was confirmed by means of the HTR Method: the average LET evaluated with TLD-700 is almost constant over the entire volume of the phantom, whereas the increasing LET measured with TLD-600 indicates an increasing flux of thermal and epithermal neutrons in the depth of the sphere. The dose equivalent distribution shows no “hot spot”. Consequently, inside a spacecraft the skin dose equivalent can be used as a reliable estimate of the risk-related whole-body effective dose.

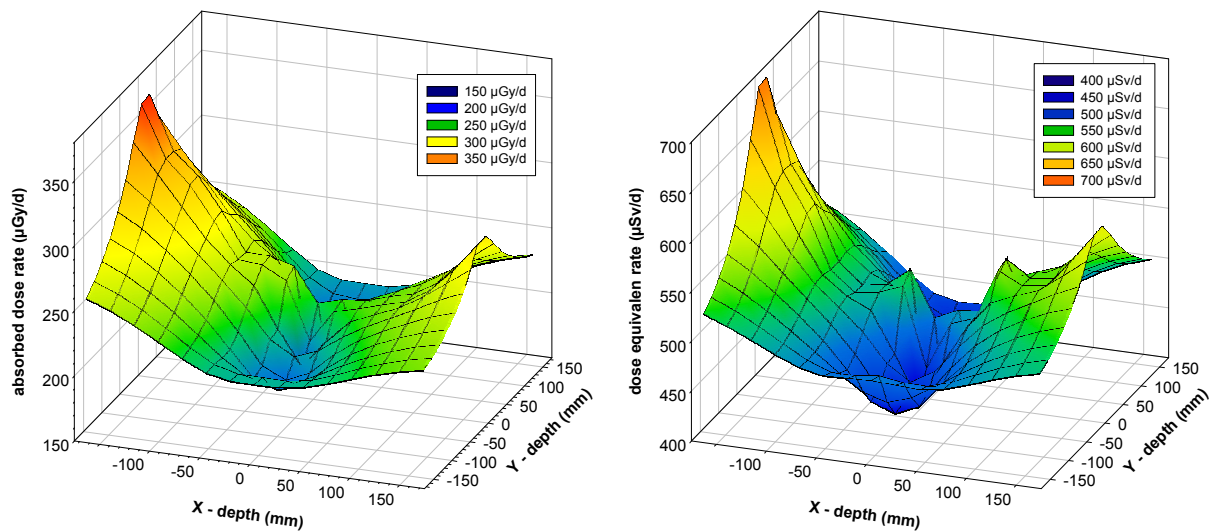


Figure 1. Depth distribution of absorbed dose (left) and dose equivalent (right) measured with TLD-700 detectors.

### Project “RBO-2 Austria” Onboard the International Space Station

The project “RBO-2 Austria” – shortly termed “BRADOS” – was started in February 2001 and comprised up to now two long-term exposures of 248 and 233 days duration onboard the ISS. A third phase is currently under progress. The detector packages containing TLD-200 (CaF<sub>2</sub>: Dy), TLD-300 (CaF<sub>2</sub>: Tm), TLD-600, TLD-700, and TLD-700H (LiF: Mg, Cu, P) were installed at six differently shielded locations inside the Russian Segment (RS-ISS). Thereby it was possible to analyze the influence of the different shielding masses on the energy and LET spectra inside the spacecraft. The highest dose rates were observed for the Commander Cabin ( $\dot{D} = 297.5 \mu\text{Gy/d}$ ,  $\dot{H} = 613.14 \mu\text{Sv/d}$ ) during BRADOS-1 and for the Core Block RS-ISS ( $\dot{D} = 226.25 \mu\text{Gy/d}$ ,  $\dot{H} = 444.16 \mu\text{Sv/d}$ ) during BRADOS-2. The lowest dose rates were found for locations near the board toilette (BRADOS-1:  $\dot{D} = 205.7 \mu\text{Gy/d}$ ,  $\dot{H} = 425.19 \mu\text{Sv/d}$ ; BRADOS-2:  $\dot{D} = 183.11 \mu\text{Gy/d}$ ,  $\dot{H} = 378.12 \mu\text{Sv/d}$ ). A similar behaviour is known also from Mir. The generally lower dose rates for BRADOS-2 stem from the unstable and on average lower ISS orbit in the year 2002.

## Comparison of Mir and ISS Results

Absorbed dose rates obtained in the Commander and Board Engineer Cabins onboard Mir and the ISS are compared in Table 1. The dose rates onboard the ISS are generally lower, although the orbital parameters are almost identical. This is partially addressed to the different solar activity conditions. The Phantom-1/2 experiments were performed in the timeframe from May 1997 to August 1998, whereas BRADOS-1 was conducted February to November 2001. On the other hand, the absorbed dose rates depend on the different shielding properties of the spacecraft. The difference between the TLD-600 and TLD-700 TL signals confirms that the neutron contribution onboard Mir is enhanced.

Table 1. Comparison of absorbed dose rates onboard Mir and the ISS.

Mir (Phantom-1/2)		ISS (BRADOS-1)	
Absorbed dose rate $\dot{D}$ ( $\mu\text{Gy/d}$ )		Absorbed dose rate $\dot{D}$ ( $\mu\text{Gy/d}$ )	
Commander cabin:	359 $\pm$ 18 (TLD 600)	Commander cabin:	302 $\pm$ 4 (TLD 600)
	332 $\pm$ 10 (TLD 700)		292 $\pm$ 5 (TLD 700)
Board engineer cabin:	385 $\pm$ 15 (TLD 600)	Board engineer cabin:	259 $\pm$ 9 (TLD 600)
	360 $\pm$ 11 (TLD 700)		248 $\pm$ 3 (TLD 700)

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# TISSUE DOSE CONVERSION FACTORS FOR PROTONS AND ALPHA PARTICLES IN CASE OF DIFFERENT DETECTOR MATERIALS

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## Introduction

One of the many risks of long duration space flights is the dose burden from cosmic radiation, especially during periods of intensive solar activity. Since the radiation field in space is a mixture of different particles differing in energy and varies with time it is difficult or all but impossible to calculate doses from earlier measurements.

There are three main methods for the determination of the astronauts' radiation exposure:

- Accumulated dose and dose-rate can be calculated from the information delivered on-line by active devices used as area dosimeters. Ionization chambers, scintillation counters, tissue equivalent proportional counters (TEPC) and silicon detector systems are widely used for these measurements.
- Active personal dosimeters such as small size silicon detectors or ionisation chambers could be used for on-line measurements. Such devices need battery power and are difficult to design small enough.
- Passive integrating detector systems such as thermoluminescent detectors (TLDs) are commonly used for environmental monitoring and for personal dosimetry, for medical dosimetry and for dosimetry in nuclear facilities, as well. Such TLD measurements need to be supported by spectroscopic information about the high Linear Energy Transfer (LET) part of the radiation field from other instrumentation (e.g. solid state track detectors) for the correction of the TLD response at high LET values and determination of the radiation weighting factor..

The basic quantity for radiation protection purposes is the equivalent dose, which is obtained by the multiplication of the absorbed dose by the radiation weighting factor ( $w_R$ ). The  $w_R$  is a function of the LET value of the concerned radiation.

The LET of charged particles in a medium is defined as the quotient of  $dE$  by  $dx$ , where  $dE$  is the average energy imparted locally to the medium by a charged particle of specified energy in traversing a distance  $dx$ . The radiation weighting factor ( $w_R$ ) is a function of LET, as it is defined by the ICRP 60. The LET value of the space radiation shows wide range distribution between 0.3 and 200 keV/ $\mu\text{m}$ .

Proportional counters and silicon detector telescopes are widely used for the measurement of the LET spectra. The main advantages of the silicon telescopes are the long-term stability and the high signal to noise ratio (i.e. suitability for measuring relativistic particles). However the silicon detectors measure the absorbed energy in the crystal instead of the LET spectra and the absorbed energy depends on the angle of incidence (i.e. the path-length) of a given particle. and the LET values of the particles are also changing in the crystal due to the change of its energy,

therefore complicated spectrum evaluation procedure should be used. Also should be taken into consideration that some particles are stopped inside the silicon crystal.

## **Space Radiation**

The usual grouping of the charged particle components of the space radiation is the following: geomagnetically trapped radiation, solar-particle radiation and galactic cosmic rays.

### Trapped radiation

As a result of the interaction of particles coming from outer space or the Sun with the geomagnetic field, there are two belts of trapped radiation, called “Van Allen belts”, surrounding the Earth as rings in the plane of the geomagnetic equator. Mostly electrons and protons are present in both belts. These particles gyrate and bounce along magnetic-field lines and are reflected back and forth between the two poles, acting as mirror. At the same time, because of their charge, electrons drift eastward, while protons and heavy ions drift westward. Electrons reach energies of up to 7 MeV and protons up to 600 MeV.

For the majority of space missions in lower Earth orbits, protons make the dominant contribution to the radiation burden inside space vehicles. At lower shielding (in the case of extravehicular activity) the total absorbed dose will be dominated by the electron contribution. Of special importance for low Earth orbits is the so-called “South Atlantic Anomaly” (SAA), where the radiation belt reaches down to altitudes of 200 km. This behavior reflects the displacement of the axis of the geomagnetic (dipole) field with respect to the axis of the geoid, with a corresponding distortion of the magnetic field. This region accounts for the dominant fraction of total exposure in ISS, although traversing the anomaly takes less than about 15 minutes and occupies less than 10% of the total time in orbit.

### Solar-particle radiation

High-energy and high intensity solar protons and heavy ions, which are among the most severe hazards for manned space flights, are emitted sporadically during solar-particle events (flares). Flares are observed mainly during the solar-maximum phase. The spectra of solar-flare protons have a large variability in absolute intensity as well as in shape.

### Galactic cosmic rays

Galactic cosmic rays (GCR) are charged particles that originate from sources beyond solar system. The GCR spectrum consists of 98% protons and heavier ions and 2% of electrons and positrons. The ion component is composed of 87% protons, 12% of alpha particles and the remaining 1% heavy ions.

The flux of GCR is affected by the sun’s eleven year cycle. During that period of the solar cycle called Solar Maximum, when solar activity is most intense, the solar wind attenuates a greater flux of the inbound GCR than during Solar Minimum, when solar activity is least intense.

GCR, being composed of charged particles, is also affected by the Earth’s magnetic field. Since the geomagnetic field lines are parallel to the Earth’s surface around the equator, all but the most energetic particles are deflected away. The geomagnetic field over the North and South Poles points towards the Earth’s surface and GCR particles of all energies are funneled toward the poles at high latitudes. The 51.56 orbit of the ISS is sufficiently highly inclined to receive a substantial exposure from less energetic GCR [1].

### Neutrons

High-energy secondary neutrons produced by interactions of high-energy charged particles (mostly galactic origin) contribute a significant fraction of the total dose equivalent in large human spacecraft as the International Space Station (ISS). The two basic components of the

neutron radiation are the albedo neutrons emanating from the Earth's atmosphere and the secondary neutrons from the interaction of high-energy space radiation with spacecraft materials. The neutron energy range of interest for radiation risk assessment is 0.1 to at least 200 MeV. Based on both the modelling results and a few measurements covering a portion of the energy range of interest, it was found that secondary neutrons contribute a minimum of additional 30 percent and up to 60 percent of the dose equivalent rates of charged particles.

### Space dosimetry

The space radiation mainly consists of charged heavy particles (protons, alpha and heavier particles) and the equivalent dose significantly differs from the absorbed dose therefore the recently used measuring equipment is not fully suitable to measure both quantities simultaneously. A new combined device consisting of the PILLE TLD system and a 3D silicon LET spectrometer is under development in the KFKI Atomic Energy Institute, Budapest [2].

The combined device in automatic readout mode will be able to determine the absorbed dose and the dose equivalent of the space radiation. This complex system will be applicable for on board calibration of the TL dosimeters measuring the effective LET value based on the analysis of the glow curves.

Because the efficiency of the TL dosimeter is a function of the LET, the values of the absorbed dose should be corrected.

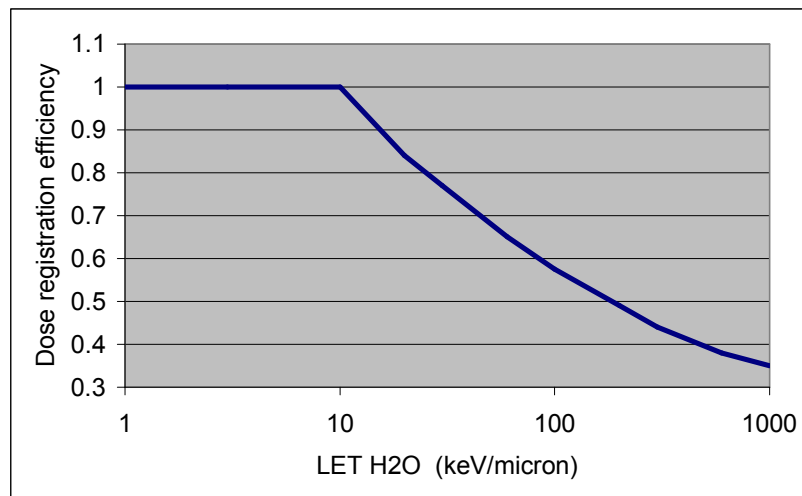


Figure 1: LET dependence of the TL-efficiency

LET-dependent Quality Factor defined by the ICRP should be used for the evaluation of the equivalent dose as well.

Since we are interested in the equivalent dose in tissue, further correction is necessary. Recently for this correction a common, LET spectrum independent value is used, despite the fact that the value of the correction factor depends on the type and energy (consequently on linear energy transfer value) of the concerning radiation, so the energy spectrum dependence of the correction factor should have to be taken into consideration for correct results.

### Databases for the linear energy transfer and stopping power

Data used for the calculations are coming from widely used databases, like ICRU 49 and SRIM, MCNP codes. As it can be seen in figure 1, the differences between these databases are not significant above 1 MeV. The ratio of the electronic and the nuclear stopping power for proton and alpha particles was also studied, it was established that above 1 MeV the nuclear collision is negligible.

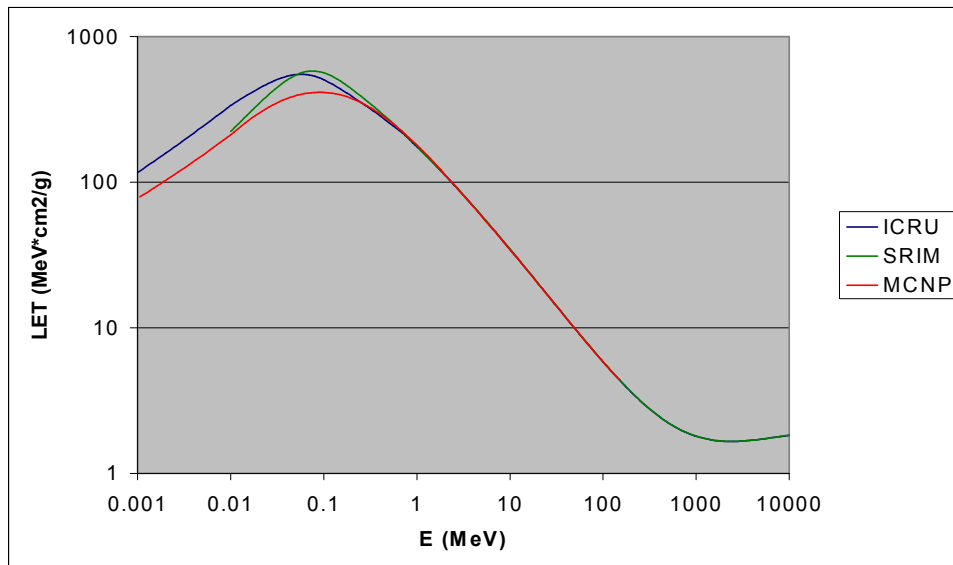


Figure 2: LET values for proton in silicon target in different databases

### Compound Correction

The linear energy transfer in compounds can be calculated as the weighted factor of the elements to a first approximation, but the stopping of ions in compounds may be different from the weighted stopping of ions in the elemental matter which makes up the compounds.

The difference in energy loss could be approximated with the “Core and Bond” approximation. This approach separates the stopping contributions from the binding electrons (bonds) from the inner shell electrons (core) [3]. If the ion has energy, E (keV), and mass, M (atomic mass unit – amu), then E/M is the ion energy in keV/amu and the following corrections could be used below 1 MeV/amu:

$$Correction = 1 + (CC - 1) / (1 + Exp(1.48 * Sqr((E/M)/25) - 7))$$

where CC is the Compound Correction. If it is equal to 1, than we don't use any correction.

If CC=1,1, for example, the stopping will be increased by 10% for ion energies of 125 keV/amu, which is the energy of the maximum stopping of protons. This correction will decrease stopping at higher energies, becoming a zero correction at 1 MeV/amu, and the correction will increase at lower energies, following the empirical formula.

### Conversion factors in case of different detector materials

During the measurements silicon spectrometers, TL materials and other detector materials are used and the energy absorbed in the silicon crystal, the light in the TL materials, etc. are measured. Since we are interested in the energy absorbed in tissue, the results have to be corrected. Correction needed for the comparison of the different measurements as well. Recently an average value is used, for example between silicon and water 1,21 [4] and 1,93 [5] are proposed.

Since the mean value depends on the energy of the incident particle, energy dependence should be taken into consideration if correct results are wanted. In figure 3 the LET values can be seen in water, calcium-fluoride, silicon, air and tissue for protons, in figure 4 the quotient between tissue and silicon is delineated. It can be seen that the average value is larger than 1 and the values change between 1,1 and 1,5.

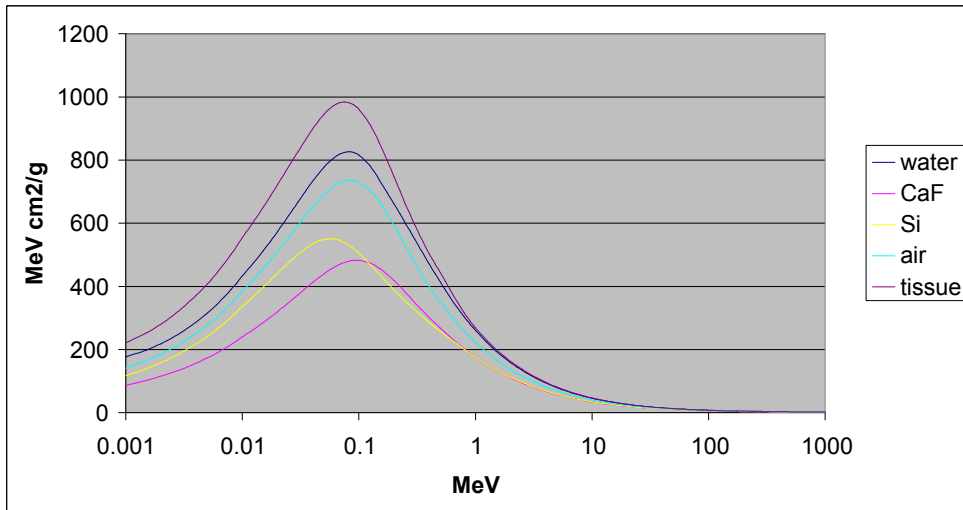


Figure 3: Proton LET values for some target

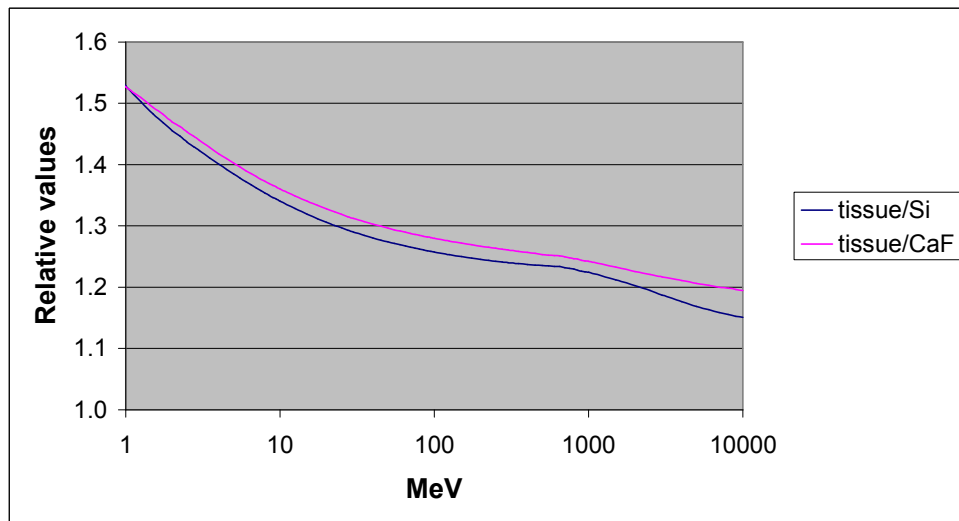


Figure 4: The quotient of the LET values for protons

Because of the reasons mentioned above the use of the average value is not recommended and the LET spectra of the incident particles should be taken into consideration. A typical LET spectrum for GCR particles at low earth orbit calculated by Atwell can be seen in figure 5 [6].

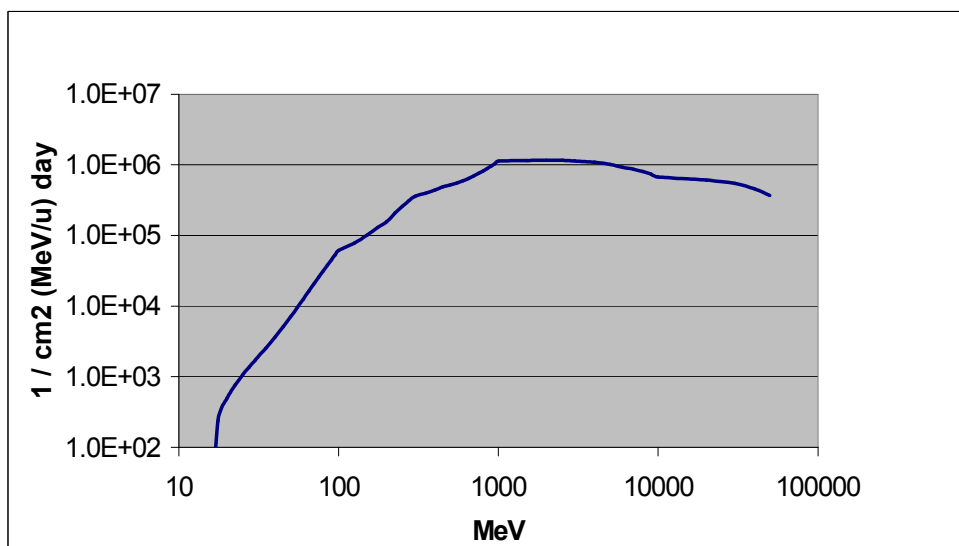


Figure 5: Typical LET spectrum for galactic cosmic rays [6]

The mean values of the correction factor between the targets could be seen in table 1. The quotients will be different if a weighted average considering the energy (or LET) spectrum is used. For silicon and tissue 1,23 could be used, while without the LET correction 1,18 is derived.

		tissue/ calcium-fluoride	water/ calcium-fluoride	water/ tissue	tissue / silicon	water /silicon
without	considering the	1.22	1.22	1.01	1.18	1.19
with	energy spectra	1.26	1.26	1.00	1.23	1.23

Table 1: Correction factors for protons (with and without considering the energy spectra)

### Conversion factors for heavy ions

Since the radiation field in space is a mixture of different particles, alpha and heavy ions should also be taken into consideration. In these cases considering the LET spectra of the particles are not so fundamental and the difference between the results is not more than 1%.

However, the weighted average for alpha particles is 1,327 which is a remarkable difference from the value calculated for protons. If the radiation field consists of different particles, these values should be averaged on the basis of the relative incidence or the ratio of the doses.

### Conclusions

Accurate dose measurement as well as the measurement of the linear energy transfer (LET) data will become increasingly important in forthcoming years during the operation of the International Space Station.

Since the space radiation mainly consists of heavy charged particles (protons and heavier particles), the equivalent dose significantly differs from the absorbed dose. While the recently used measuring equipment is not fully suitable to measure both quantities simultaneously, a new combined device is under development.

The efficiency of the TL dosimeter is a function of the LET, therefore the value of the absorbed dose should be corrected. LET spectra or mean LET value should be used for the determination of the radiation weighting factor ( $w_R$ ) and for the evaluation of the equivalent dose as well. Since the radiation weighting factor defined as a function of  $LET_{water}$ ,  $LET_{Si}$  or  $LET_{TL}$  should be converted to  $LET_{water}$  for this purpose. Correction needed for the calculation of the dose in tissue and for the comparison of the different measurements as well.

Recently an average value has been used for the correction of the LET between different materials, despite the fact that the value of the correction factor depends on the energy of the concerning radiation. The discrepancy of the mean values with and without considering the energy spectrum could be more than 5%.

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# LINEARITY AND LET SENSITIVITY OF LiF TL DOSIMETERS

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## Introduction

Manufacturers as well as users devote much effort to studying the dosimetric characteristics of TLD-100 (Harshaw) and MTS-N (Poland) LiF dosimeters, LiF(P). These dosimeters are widely used and easily available for our every day practice in the field of work place monitoring to measure low LET (gamma) and mixed low and high LET (eg. gamma-neutron) radiation.

The LET dependence of TL dosimeters has been investigated by many authors (1-2). Results indicate that for all commonly used materials the sensitivity (TL response/absorbed dose) decreases with increasing LET. However, there are a few exceptions, such as an increase in the sensitivity of the 240-270°C peaks of LiF (referred to as peak 6-7). The 'low' temperature (180-210°C) peak of LiF (referred to as peak 4-5) decreases with increasing LET.

There is no harmony information in the literature on the linearity of LTP (low temperature peak) and HTP (high temperature peak) using high LET radiation field, because in the most cases the gamma background of neutron radiation does not take into consideration by a control dosimeter having low neutron sensitivity.

In our earlier paper (3) the selectivity of different LiF dosimeters (TLD-100, MTS-N) have been compared under the same conditions and the variation of the peak temperature with heating rate were studied.

The aim of this paper was to investigate the linearity of LTP and HTP for both LiF and to compare the LET sensitivity of TLD-100 (Harshaw) and MTS-N (Poland) LiF personal dosimeters using thermalized neutron radiation in the range of 1 to 150 mSv dose.

## Materials and method

The experiments were performed with <sup>137</sup>Cs gamma and <sup>238</sup>Pu-Be neutron source (flux:  $2 \times 10^7$  n/s) with a 20 cm diameter spherical polyethylene moderator.

The various dosimeters investigated were as follows:

- LiF(P) from Poland (type: MTS-N) (4), size diam. 4.5x0.7 mm, sintered, weight: 20 mg,
- TLD-100 from Harshaw, size 3.1x 3.1x0.8 mm<sup>3</sup>, hot pressed, weight: 24 mg,

Al<sub>2</sub>O<sub>3</sub>:Mg,Y ceramic TL dosimeters (size diam. 8mm x 1 mm) were used to control the gamma background of the neutron irradiation, having low sensitivity to thermal and fast neutrons of energies up to 4 MeV (5). The measured gamma background was subtracted from the TL response of the neutron irradiation.

The LiF pellets and the Al<sub>2</sub>O<sub>3</sub> dosimeters were read out one day after the irradiation using a Harshaw 2000 AB TLD reader with evaluation software on an IBM compatible computer. The temperature of heating cycle reached 300°C during the evaluation with heating rate of 5°C/s.

Annealing was made between irradiations:

- 1 hour at 400°C and 2 hours at 100°C for both LiF dosimeters.
- 1 h at 600°C for Al<sub>2</sub>O<sub>3</sub> dosimeters

During thermal neutron irradiation by <sup>238</sup>Pu-Be source the dosimeters were placed inside a polyethylene moderator.

## Results and discussion

Five pieces of each type of dosimeter were irradiated first with a gamma dose of 10 mGy and then with neutron dose of 15, 75 and 150 mGy, respectively. The dosimeters were read out one day after the irradiation. The irradiated dosimeters were evaluated twice to obtain the background reading as well.

The thermoluminescence glow curves of LiF(P), TLD-100 and Al<sub>2</sub>O<sub>3</sub>:Mg,Y dosimeters irradiated with gamma dose of 10 mGy are shown on Figure 1. The gamma sensitivity of LiF(P) was found 2.5 times higher comparing TLD-100 which is in good agreement with the literature data.

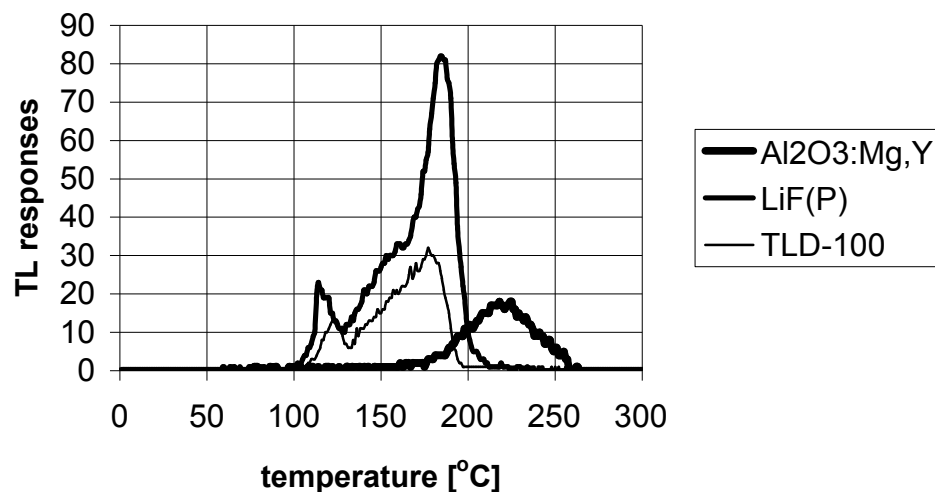


Fig. 1. Thermoluminescence glow curves of TLD-100 and LiF(P) TL dosimeters at 10 mGy of gamma radiation

The TL glow curves of LiF(P) and TLD-100 dosimeters irradiated by 15 mSv and 75 mSv thermal neutrons can be seen on Fig. 2 and Fig. 3. The TL response of the Al<sub>2</sub>O<sub>3</sub>:Mg,Y was negligible after the neutron dose of 15 mSv irradiation.

Increasing the dose of neutron irradiation by a factor of 5 and 10, there is further growth of high temperature peak and a substantial increase in the width of low peak.

The TL light outputs measured under the peaks of dosimeters irradiated with neutrons indicate that the high peak at 240°C temperature is linear for both LiF dosimeters in the dose range of (1-150 mSv) investigated (Fig. 4).

Another result that is worth mentioning is related to the gamma and neutron sensitivity of dosimeters investigated. Despite of the very different gamma sensitivity of LiF(P) and TLD-100 dosimeters (sensitivity of LiF(P) is about 2.5 times higher compared to TLD-100 one, there is not so much difference between their neutron sensitivities: 1.4.....1.25 using neutron dose of 15 mSv and 150 mSv , respectively.



Comparing the selectivity of glow curves of LiF(P) and TLD-100 dosimeters, the ratio of HTP (peak 6-7) to LTP (peak 4-5) was found similar (about 0.40) for both types of dosimeters in the neutron dose investigated.

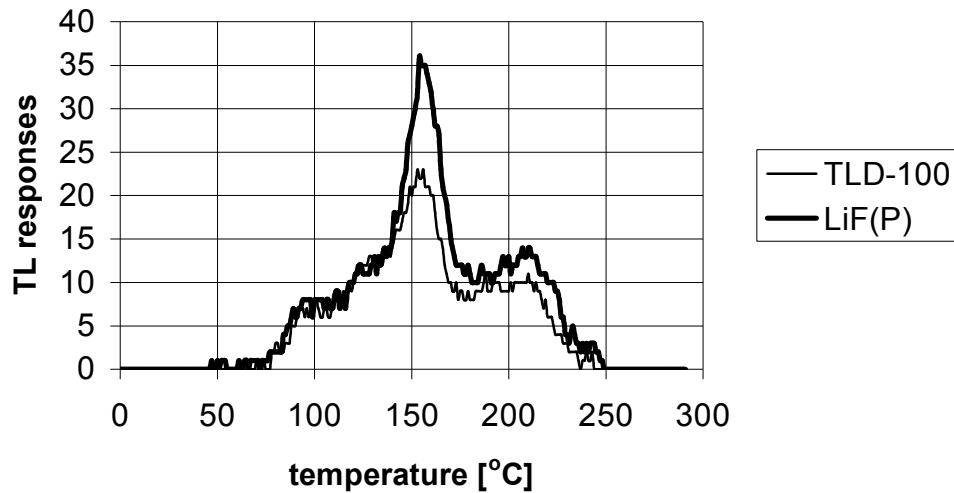


Fig. 2. Thermoluminescence glow curves of TLD-100 and LiF(P) TL with 15 mSv thermal neutron irradiation

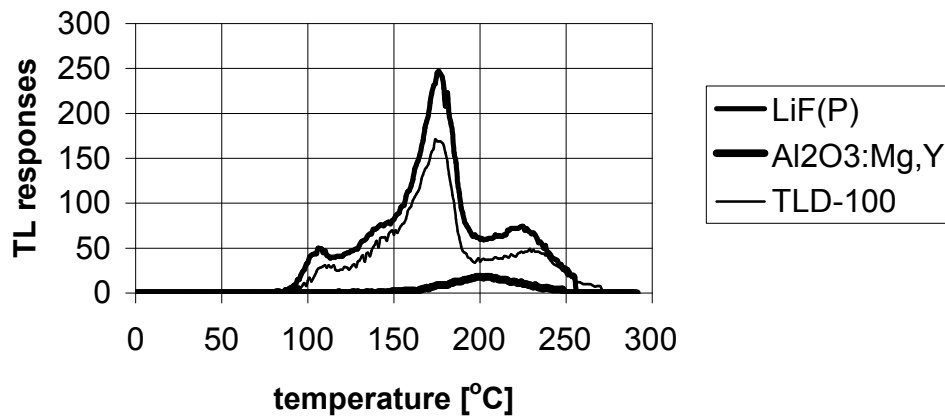


Fig 3. Thermoluminescence glow curves of TLD-100 and LiF(P) TL with 75 mSv thermal neutron + 10 mGy gamma irradiation

## Conclusion

Several investigations have been conducted during the past 30 years studying the characteristics of the most widely used TL dosimeters, namely the LiF:Mg,Ti dosimeters. There is no harmony information in the literature on its LET dependence, especially on the linearity of low and high temperature peaks using high LET radiation field.

In our paper the LET dependence of TLD-100 and LiF(P) dosimeters was investigated to determine the efficiencies of TL light production and to assess the change on the structure of glow curves by increasing dose of neutron radiation.

It was found, that while the gamma sensitivity of LiF(P) to TLD-100 is about 2.5, the neutron sensitivity differs not so much, i.e. 1.4 (at 15 mSv) and 1.25 (at 150 mSv). The high

temperature TL peak is linear for both kinds of LiF dosimeters in case of the thermal neutron dose range investigated.

The high temperature TL peak is linear for both kinds of LiF dosimeters in the thermal dose range (1–150 mSv) investigated.

In a respect of selectivity both LiF(P) and TLD-100 are suitable for selective measurement of neutron dose in a mixed neutron-gamma radiation for accidental dosimetry purposes.

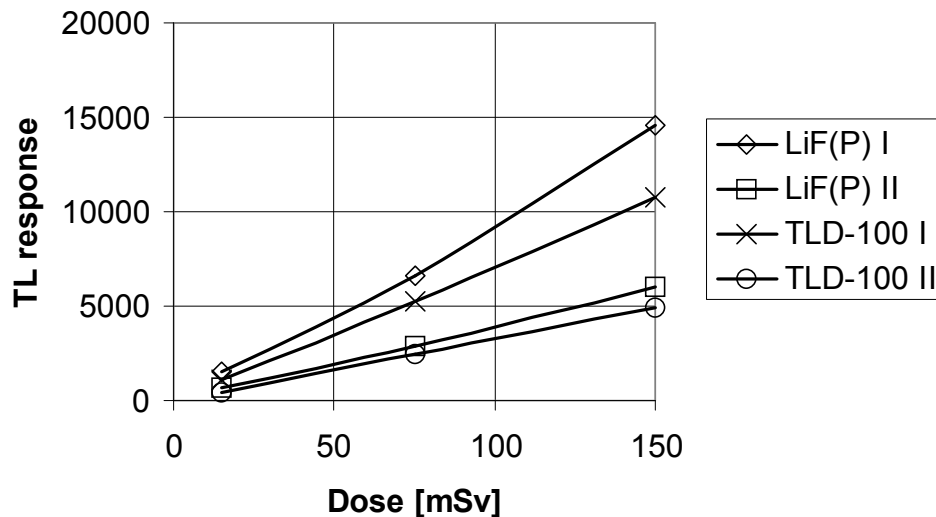


Fig. 4. Linearity of TL responses of TLD-100 and LiF(P) TL dosimeters using their low temperature peaks (LTP) and high temperature peaks (HTP)

### Acknowledgement

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# **PERSONAL DOSIMETRIC MONITORING IN UKRAINE: CURRENT STATUS AND FURTHER DEVELOPMENT**

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## **Current status of dosimetric monitoring in Ukraine**

In Ukraine the number of individuals exposed occupationally, medically or by residence in the areas with technogenically enhanced levels of radiation is scoring tens of thousands. Unfortunately, individual dosimetry is performed by uncoordinated laboratories, using obsolete equipment and practicing inadequate measurement techniques, there is no centralized infrastructure for concentration and dissemination of information on individual doses. Aforesaid calls for elaboration of the nationwide System for management and coordination of dosimetric monitoring as well as deep modernization of existing dosimetry services.

Presently Ukraine has mixed system for dosimetric monitoring. Nuclear power plants and some major nuclear facilities have their own dosimetry services, which are responsible for regular dosimetric monitoring of workers. Rest of occupationally exposed persons is monitored by dosimetry laboratories affiliated to the territorial authorities for sanitary and epidemiology supervision.

In 2002-2003 Ukrainian Ministry of Health performed survey of the status of dosimetric monitoring and inventory of critical groups requiring such monitoring. Dosimetry services in Ukraine cover about 38,000 occupationally exposed workers, including 9,100 medical professionals, 16,400 employees of 5 nuclear power plants and ca.12,400 workers dealing with other sources of occupational exposure (industry, research) (Figure 1). Territorial dosimetry services operate in 13 of 24 oblasts of Ukraine, using DTU-01 manual TLD readers produced with one exception in 1988-1990. The coverage of critical groups by dosimetric monitoring is variable and ranges from 38% to 100% depending on the oblast.

Personnel of nuclear power plants (about 16,400 workers) is monitored by their own

dosimetry services achieving absolute coverage of the main staff and temporary workers. NPPs IDM instrumentation and covering is shown in Table 1

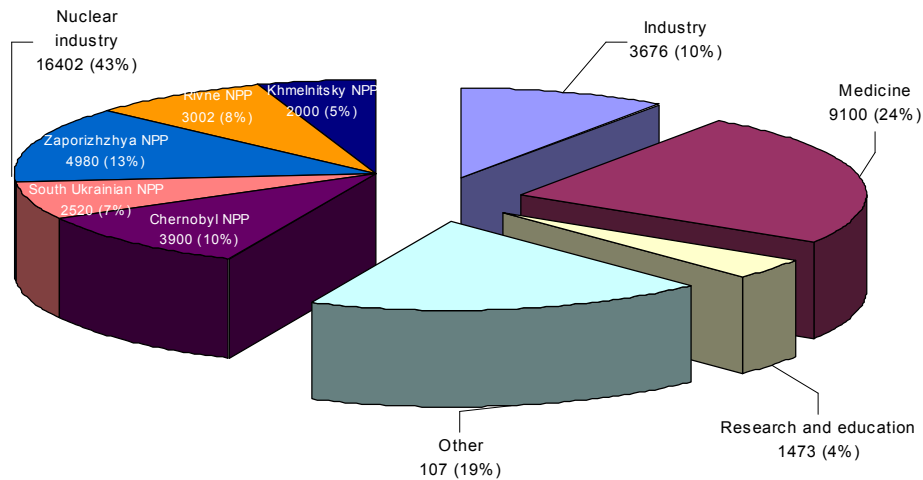


Fig.1. Structure of Ukraine Dosimetry services coverage

Table 1. Nuclear power plants dosimetry services

Nuclear power plant	Instrumentation	Year of installation	Number of dosimeters	IDM covered persons
Chernobyl NPP	Harshaw 8800	2000	-	3900
South Ukrainian NPP	КДТ-02М	1983-1991	3600	2520
Zaporizhzhya NPP	КДТ-02М	1983-1994	13000	4980
Rivne NPP	АКІДК-101	1994	8000	3002
Khmelnytsky NPP	КДТ-02М	1985-1998	1194	2000

Instrumentation used for monitoring of nuclear workers is different: Chernobyl NPP operates new Harshaw 8800 TLD systems, semi-automated TLD system AKIDK-101 (Russia) is used in Ryvne NPP; personnel of three other NPP is monitored using manual TLD readers KDT-02M. Each of five NPP has its own data environment.

Dosimetric monitoring of medical personnel is mainly performed from the single service - Central laboratory of radiation hygiene of medical personnel affiliated to the Grigoriev Institute of Medical Radiology AMS Ukraine in Kharkiv. Totally, 5,500 medical staff members are monitored by this service on quarterly basis using obsolete manual TLD readers DTU-01.

## Elaboration of the united system

The System is intended to cover all aspects of efficient dosimetric monitoring, in particular-provision of methodical unity of individual dosimetric monitoring, scientific and methodological guidance of individual dosimetric control, procurement of common technical policy regarding nomenclature and operation of instrumentation, implementation of quality assurance programs, development and support of common information infrastructure intended for logging, storage and access to data on individual dosimetric monitoring as well as keeping the national registry of individual doses, training and certification of personnel engaged in the System of individual dosimetric monitoring. Three levels of the System (Figure 2, Table 2) have different goals and responsibilities.

Table 2. Tasks of the Levels of the proposed System

<b>Level</b>	<b>Tasks</b>
<b>Level 1</b> (Central IDM lab, dosimetry registry and coordination centre)	development of the guidelines in the area of dosimetric monitoring and data handling; provision of methodological support for operation of the System; development and implementation of the unified technical policy; development of the information infrastructure; analysis and dissemination of data regarding the results of dosimetric monitoring; training and quality assurance for the local dosimetric monitoring facilities.
<b>Level 2</b> (Regional IDM services)	maintain regional dosimetry registries; concentrate the results of dosimetric monitoring performed in the region of their responsibility by all dosimetry services incorporated into the System; take over routing dosimetric monitoring in respective territories (later, when laboratories of the second level will equipped with modern equipment).
<b>Level 3 (a)</b> (well equipped dosimetry laboratories of the NPPs)	routine dosimetric monitoring of the personnel; dosimetry database management and keeping raw data of dose measurement; transfer of the results of dosimetric monitoring to the State dosimetry registry.
<b>Level 3 (b)</b> (territorial sanitary and epidemiology stations)	distribution and collection of dosimeters, which are issued by the laboratories of the first and second levels; supervision of the usage of dosimeters; feedback to end users – transfer of the results of monitoring to persons and management; maintaining of the local level of the State dosimetry registry.
<b>Level 3 (c)</b> (independent services after proper accreditation)	dosimetric monitoring on commercial basis following methodological guidance provided by the first level of the System; keeping of the raw data of dose measurement; transfer of the results of dosimetric monitoring to the State dosimetry registry.

The process of development of the System will have several stages. At the first stage information infrastructure of the System should be established and measures will be taken for harmonization of the measurement techniques. At the second stage well equipped central laboratory will be established eliminating inefficient small dosimetry services. At the third stage regional laboratories (Level 2) will be established and equipped with modern instrumentation.

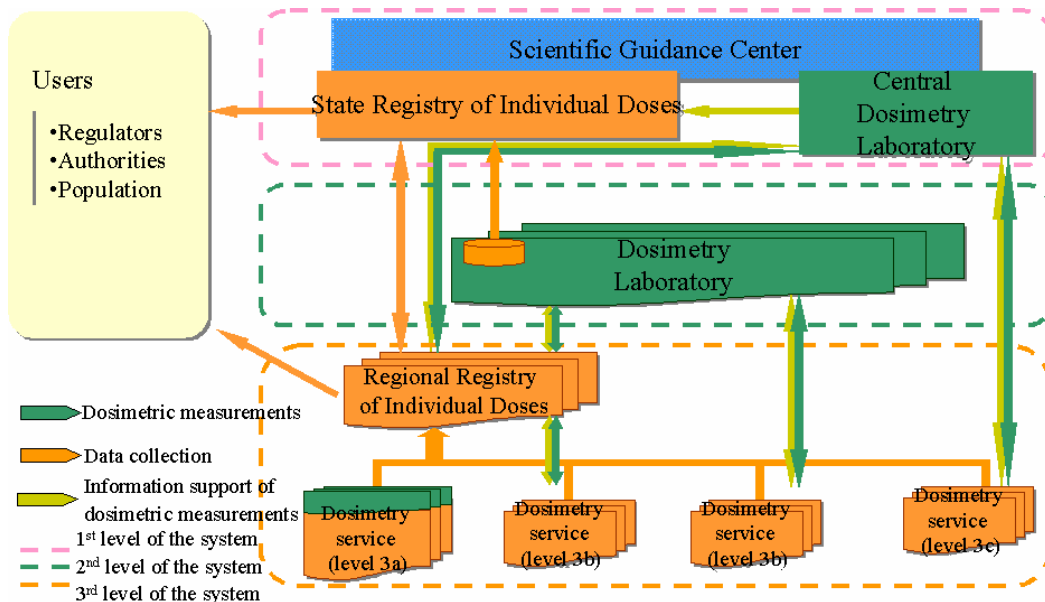


Fig.2. Structure and operation flowchart of the unified state system for monitoring and registration of individual doses of population

## Conclusions

Current inadequate status of dosimetric monitoring infrastructure in Ukraine demands an urgent elaboration of the united state system for monitoring and recording of individual doses. The proposed plan would allow to bring dosimetry infrastructure in Ukraine to the modern state which would be compatible with existing and future European and international radiation protection networks.

Unitary structure of Ukraine, strong administrative command and good communications between regions of the country are positive factors in favour of efficient implementation of the proposed plan. Deficiencies are associated with limited funding of this effort.

# **ESTIMATION OF INTERNAL EXPOSURE FOR EXPOSED PERSONEEL FROM A CANDU NUCLEAR FUEL FACTORY**

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## **ABSTRACT**

The knowledge of the radioactive material behavior inside the human body is an essential issue for interpretation of the radioactivity measurements in human body or excretions in terms of internal contamination or committed equivalent dose.

The paper present evaluation of internal contamination of professionally exposed workers from a CANDU nuclear fuel factory with the ACRO computer code which estimate burden and tissue or organ dose resulting from inhalation or ingestion of radioactive materials. The workplaces where continuos aerosols sampling are carried out, has been taken into account for the analysis. For potentially inhaled activity assessment, the average aerosol concentrations were estimated. The dose equivalent and collective dose equivalent are also estimated.

### **1.Air radioactivity**

Air radioactivity measurement can give a more direct estimation of the personnel exposure. In this case, determination of the radioactivity of uranium aerosols gives the opportunity to evaluate the internal contamination of exposed personnel.

The use of the data obtained by means of this method is limited to the following considerations:

- Distance between air samples and exposed person
- Collection of particles from the air filters does not provide an accurate value of the inhaled radioactivity, since the particles behavior in the respiratory tract is strongly dependent on their size. This fact must be considered in dose estimation resulted following the inhaling of radioactive isotopes.

## 2. Excretions and biological samples

Biological samples analysis can be used in uranium detection in human body. This method has the advantage that the presence of the individual is not needed.

Analytical and radiochemical methods are time-consuming and the results obtained are delayed and are not identical. The collection requires the cooperation of the person. However, the collection of such samples can be unpleasant if they are collected for a long period of the time, so that adequate radioprotection control can be done.

In the case of uranium, due to its metabolic behavior, urine can be used as a contamination indicator. The interpretation of the results of measurements of the radioactivity in urine implies a series of errors: the functions describing excretion are not well known and there are significant variations for the chemical forms with large retention periods in the body ( $UO_2$ ). The excreted activities reflect long-time accumulations and assist in the assessment of the internal contamination.

## 3. Biokinetic method

The estimation of internal radiation exposure is of considerable importance in radiological protection for both, workers and the public, because human organism or tissues may be directly exposed to radioactive materials retained in the body for substantial period.

There are two main routes through which radioactive materials are taken into the human body: one is the respiratory tract and the other is the gastrointestinal tract. The precise estimation of internal exposure requires detailed knowledge of radiological and physiological behavior of radionuclide in the body, but such estimation can not be generally applied to radiological protection. ICRP adopted a relatively simple dosimetry model and related metabolic data in which the maximum permissible concentration and maximum permissible body burden for workers were calculated. This model and data have still been widely used for radiological protection. Software programs have been developed and one of them is used in the ICRN Radiation Protection Laboratory for the calculation of internal dosimetry. In the ICRP publication the maximum permissible concentration and maximum permissible body burden for workers were calculated.

Following the progress and increase of knowledge, more precise and detailed models have been developed. The lung dynamics model developed by the ICRP task group on lung dynamics has been applied to make a more precise estimation of internal exposure following the inhalation of radioactive materials. In this model the respiratory tract is divided into four regions, i.e. NP (Naso-Pharyngeal), TB (Tracheo-Bronchial), P (Pulmonary parenchyma) and



L (pulmonary lymphatic system) regions. The lung is defined as a composite organ of TB, P and L regions.

Inhaled materials are deposited in these regions with varying probabilities depending on the particle size. Deposited materials are excreted from the regions to the body fluids and to the gastrointestinal tract (GI tract) or exhaled with different removal rates determined by solubility. The half-time is determined by solubility of the inhaled materials that are categorized into three classes by ICRP:

- class D for well-soluble (daily excretion);
- class W for soluble (weekly excretion);
- class Y for insoluble (annual excretion).

The ACRO program is written in FORTRAN IV and was obtained from NEA Data Bank from France. The test case has been obtained and also prediction capacity hosted verified for other input data. A concordance with specialized literature has been obtained.

The input data are:

- the organ for which the evaluation is desired;
- atomic number;
- atomic mass;
- solubility class of inhaled or ingestion isotope;
- inhalation or ingestion period;
- total radioactivity incorporated in this period ( $\mu\text{Ci}$ );
- the particle diameter of inhaled materials (AMAD);
- time for the dose and burden calculation for the organ or tissue n (in seconds, hours, days or years);
- the duration of contamination: ingestion or inhaling;
- human type (standard man or reference man).

#### 4. Assessment of internal exposure

Radiation protection must ensure humans protection in the case of activities involving radiation sources. Its main objectives are to prevent harmful effects on the humans health and to restrict the probability of occurrence of cancer or hereditary effects at reasonably acceptable levels. ICRP [2] has established the principles for presently accepted dose limitations in radiation protection.

Since it is difficult to measure a basic value, such as the actual dose equivalent, a derived measurable value is required [3]. In the case of internal exposure, as a result of ingestion or inhaling of radioactive substances, various organs and tissues of the human body will not be uniformly irradiated, depending on several factors: the fraction of the incorporated amount which affects the organ/tissue, the disintegration constant of the inhaled radionuclide, its physical and chemical properties, etc. The actual dose equivalent will be function of all these factors and of the quantity of substance ingested.

Uranium is processed at the Nuclear Fuel Factory (FCN) under various physical and chemical forms. During processing, the exposed personnel can inhale aerosols which contain uranium, and thus the risk of internal contamination arises. This contamination results in irradiation doses on the internal organs or toxic chemical effects – if uranium is in large amount. The chemical effects are more influential than the received dose.

In order to evaluate the dose equivalent and the collective dose of FCN personnel we had to determine the potentially inhaled radioactivity ( $\mu\text{Ci}$ ) from the average value of aerosols concentration for each sampling point. The following Table presents the aerosols sampling points within FCN:

No. of sampling points-Solubility class	Activity performed
6-D	Dissolving & treatment
9-D	Precipitation
18-W	Pellets loading
24-Y	Sorting/forming of columns
28-Y	Grinding/rectification machine
30-W	Pelletizing press
32-Y	Grinding machine
36-W	Pelletizing press
38-W	Briquetting press
11; 12;15-D+W	Reduction furnace supply, powder storage
18; 5-Y	Pellets loading ; discharge table
5; 18; 25-Y	Discharge table; pellets loading; dismantling lathe
55; 58-Y	Lathe for bundles dismantling; pellets loading table
8; 10-D	Drying furnace; granulator; drying furnace entrance

For personnel working in these points the natural uranium concentration values in urine were radiochemically determined [6,7]. These values, as well as the average aerosols concentration, are presented in the Table below:

Aerosols measuring point	Average aerosols concentration (Bq/m <sup>3</sup> )	L.C.A. (Bq/m <sup>3</sup> )	Uranium average concentration in urine (µg U/l)	L.C.A. (µg U/l)
6	0.079	2.2	7.25	50
9	0.196	2.2	13.50	50
18	1.344	2.2	5.40	50
24	0.477	2.2	11.50	50
28	0.350	2.2	9.80	50
30	0.352	2.2	7.00	50
32	0.339	2.2	8.50	50
36	0.440	2.2	8.75	50
38	0.713	2.2	7.25	50
11;12;15	0.345	2.2	14.50	50
18;5	0.737	2.2	4.00	50
5;18;25	0.535	2.2	5.86	50
55;58	0.041	2.2	1.25	50
8;10	0.276	2.2	2.86	50

In the case of natural uranium dioxide powders, by means of ACRO code the burden on potentially affected organs was assessed (kidneys, bone, lung, stomach and the other components of the gastro-intestinal tract) and also the dose-equivalent on these organs.

The assessment took into account a work period of 150 days. The intervals for which the estimation was done are: 365, 1826 and 18250 days.

The equivalent dose  $H_E$  has been calculated by means of the relation:

$$H_E = H_T * W_T$$

Where:

$H_T$  = equivalent of dose received by organ T and

$W_T$  = the weighting factor of organ T

In our case, the irradiation of several organs was considered, thus:

$$H_E = H_T * W_T$$

The equivalent dose and the collective dose equivalent, corresponding to each work-post, have the following values:

Aerosols sampling point	Equivalent dose (mSv)	No. of persons surveyed	Collective dose (mean/mSv)
6	1.56E-03	8	1.25E-02
9	3.84E-03	2	7.67E-03
18	2.64E-02	10	2.64E-01
24	9.38E-03	1	9.38E-03
28	6.88E-03	5	3.44E-02
30	6.90E-03	4	2.76E-02
32	6.66E-03	2	1.33E-02
36	8.66E-03	2	1.73E-02
38	1.40E-02	5	7.01E-02
11;12;15	6.81E-03	5	3.41E-02
18;5	1.45E-02	1	1.45E-02
5;18;25	1.05E-02	7	7.36E-02
55;58	8.01E-04	2	1.60E-03
8;10	5.42E-03	7	3.80E-02

The equivalent dose represents the most important value in radiation protection. In the case of occupational exposure the regulations provide [5] an annual limit of 20 mSv.

## CONCLUSIONS

Considering the limits of the ACRO program one can state that internal exposure for the professionally exposed personnel within FCN is far below the values encountered in specialized literature. The value provided by regulations for external exposure is 20 mSv/year, so the contribution of internal exposure to the total dose is insignificant.

As it can be seen from the above table, in the case of the considered work-posts the limits provided for by international standards and rules are not exceeded.

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# NETWORK FOR COLLECTION OF TEETH FOR RETROSPECTIVE EPR DOSIMETRY IN UKRAINE

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## **Introduction**

Today EPR (Electron Paramagnetic Resonance) dosimetry with tooth enamel is well established method of individual dose reconstruction, which was brought to the level of high reproducibility and precision; the guidelines for application of EPR dosimetry were recently published in a form of IAEA TECDOC-1331 [1].

However limited availability of specimens is a serious drawback when large scale application of EPR dosimetry is concerned. The matter is that teeth could be extracted from the subjects of a study by medical prescriptions only and loss of teeth is random and infrequent event.

In Ukraine the problem of retrospective dose estimation is of particular significance because the official dosimetric information for most part of Chernobyl clean-up workers is sparse and incomplete. Besides the available dose information has poor quality and thus demands verification and validation.

Scientific Center for Radiation Medicine AMS Ukraine is in charge with the problems of retrospective assessment of Chernobyl exposures for many years. EPR dosimetry with tooth enamel is applied here since 1993 for retrospective determination of liquidator doses from external gamma-irradiation.

Elaboration of the nation-wide network for collection of dosimetric stuff was conceived in May, 1997 by the Order of the Ministry of Health of Ukraine. It was repeated by the orders on regional and city public health services, which had determined base medical institutions (predominantly regional dental off-patient clinics) and specialists responsible for the collection of liquidator's teeth in each area.

As a result of this initial effort till the end of 1997 the network for collection of a dosimetric stuff was organized in Kiev and in four oblasts (regions) of Ukraine: Dnipopetrovsk, Zaporizhye, Kharkiv and Poltava. The first dosimetric specimens were received already at the beginning of 1998 and the number of collected teeth steadily increases (Fig.1). In 2000 the tooth collection network was expanded and covered also Chernihyv and Cherkassy oblasts.

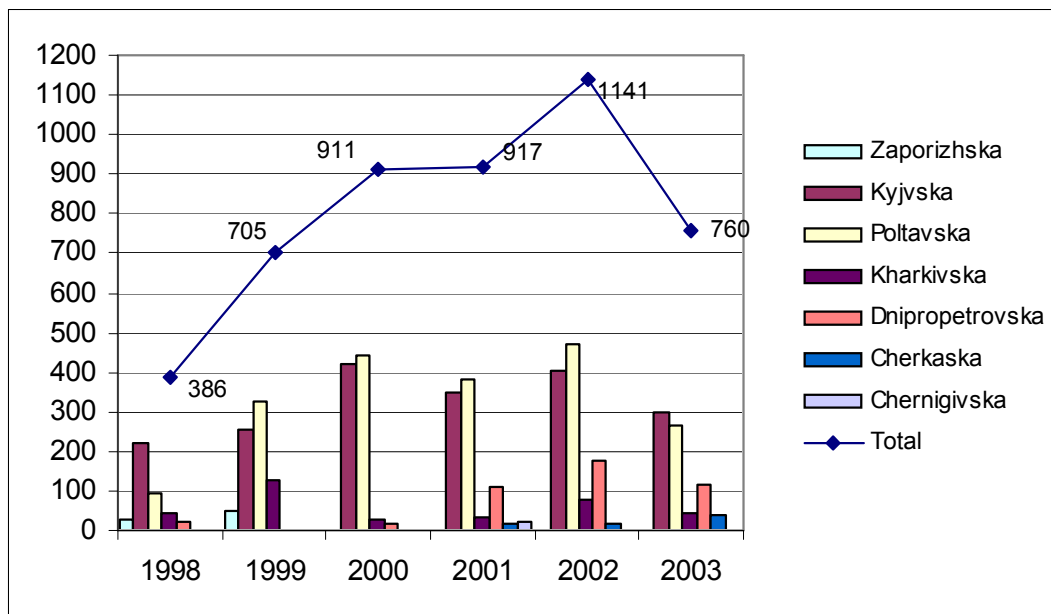


Fig.1. Dosimetric material collection dynamics. Year 2003 is not complete.

Particular role in operation of a network for collection of dosimetric specimens belongs to the central bioprobe bank, which in particular serves to retrospective estimation of individual radiation doses of the liquidators. The bioprobe bank not only accumulates and stores the dosimetric specimens but also incorporates the information about the liquidators, which is indispensable for dose reconstruction.

The flowchart describing the network and interrelation between its elements is shown in a Fig.2.

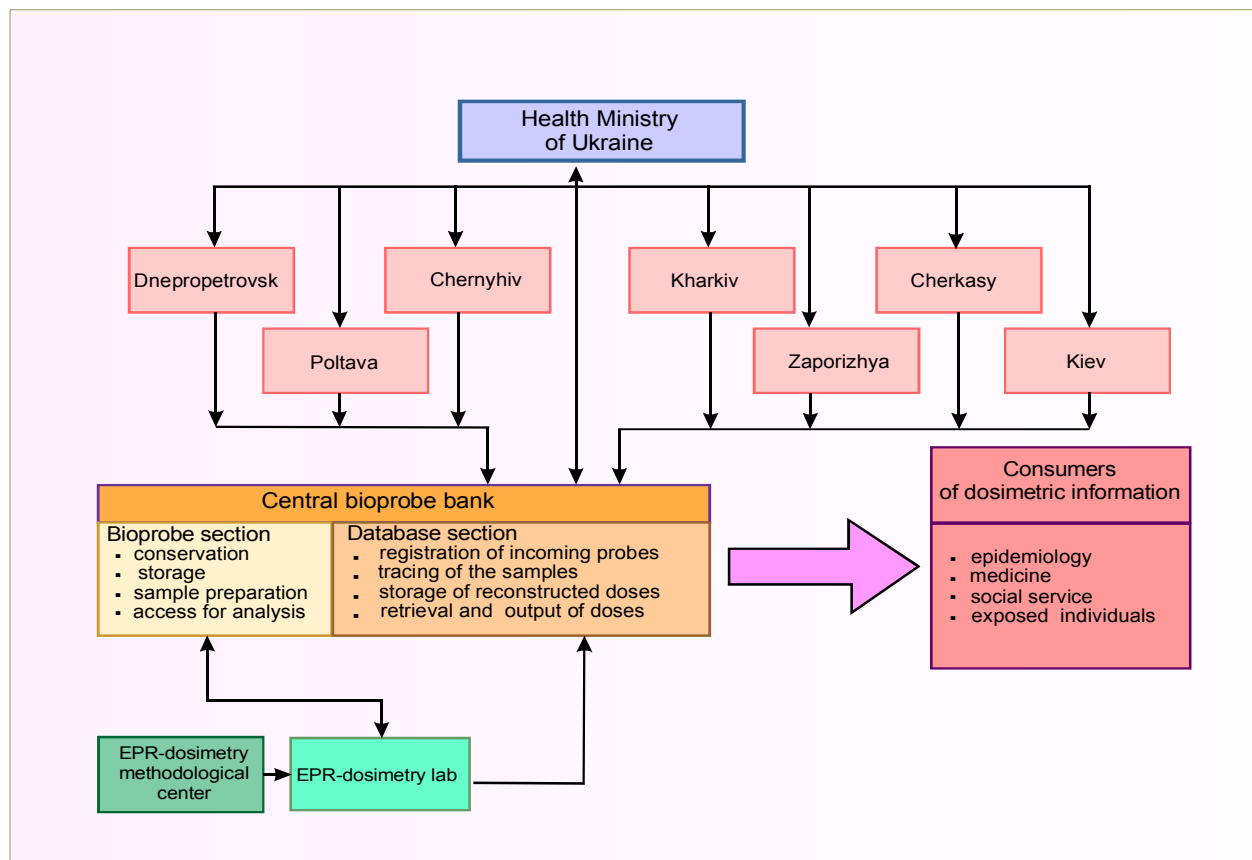


Fig.2. Flowchart of the nation-wide tooth acquisition network in Ukraine

Network for collection of teeth for retrospective EPR dosimetry in Ukraine today consists of 167 medical institutions ranging from regional dental polyclinics, city and district hospitals down to local ambulatories and medico-sanitary units; 314 dentists take part in this effort. Growth of this infrastructure is shown at Fig.3. For example, only in the Poltava area the dosimetric specimens arrive from 71 medical institutions; the teeth are collected and the accompanying documents (tooth ID forms) are being completed by 145 dentists.

Over five years of the full-scale operation of the network, 5,680 teeth from 3,574 liquidators were collected, about 2,600 teeth were found to be adequate for high precision (two aliquot) EPR dosimetry, 638 doses were reconstructed.



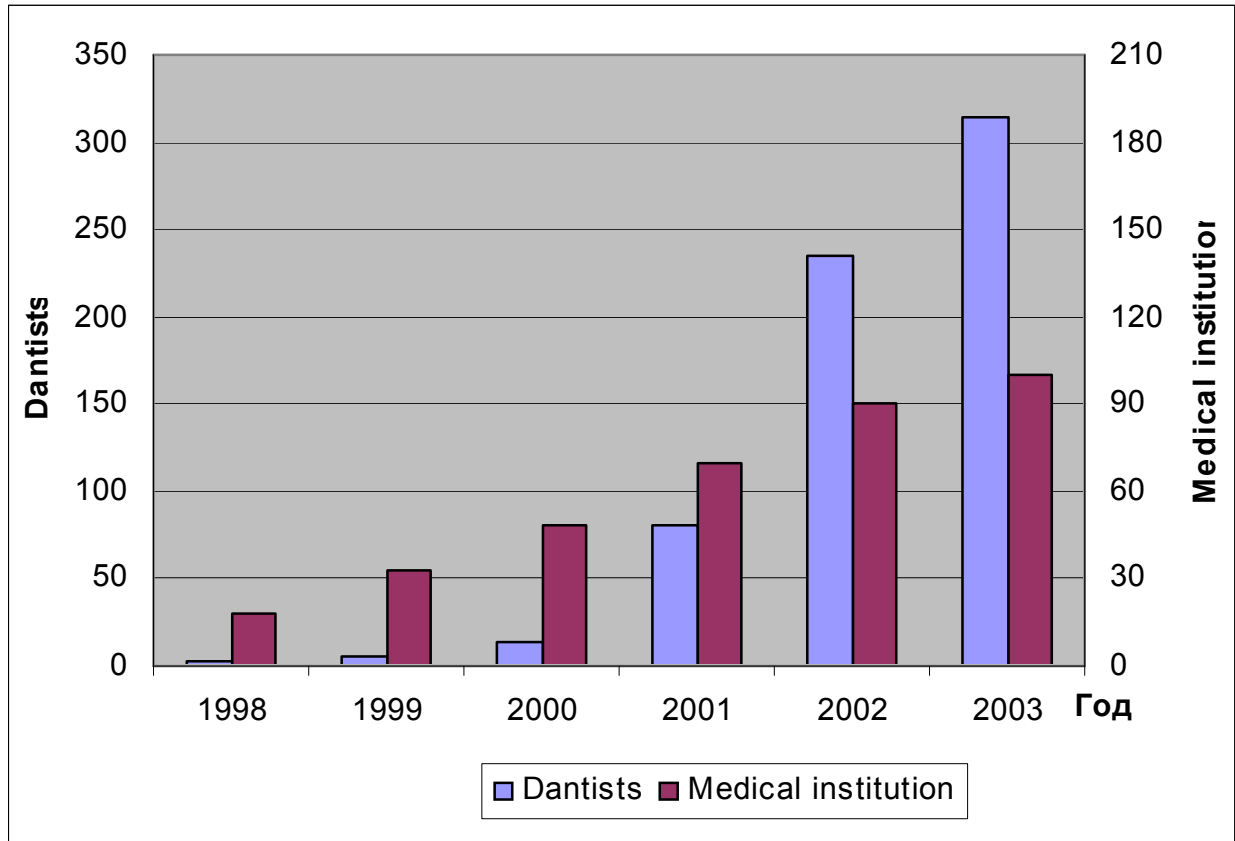


Fig.3. Growth of the tooth collection infrastructure

### Conclusions

Network for collection of liquidator’s teeth had been established in Ukraine by joint efforts of SCRM AMSU and the Ministry of Health, substantial support is being received from international projects, in particular from the German-French Initiative “Chernobyl”. It ensures the accumulation of the dosimetric specimens and provides necessary conditions for wide scale application of the retrospective EPR dosimetry with tooth enamel. The high precision doses reconstructed for Chernobyl liquidators by the means of EPR dosimetry may be used for validation of other dosimetry techniques and verification of existing dose records for variety of epidemiological applications.

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# LONG-TERM AIRCREW EXPOSURE MONITORING BY MEANS OF A SI-DIODE SPECTROMETER

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## **Introduction**

Since the publication of ICRP 60 recommendations [1], aircrew exposure has become one of most actual problems of occupational exposure. Workshops were organized [2], some review reports were published [3]. In this contribution we present the results of onboard measurement by means of a Si-diode energy deposition spectrometer, MDU-Liulin [4]. They were accumulated during 2001-2003 years onboard of an A310-300 aircraft of Czech Air lines, during 6 long-term monitorings and about 30 return single flights. Some of results obtained are presented, discussed and analyzed.

## **Materials and methods**

### *Semiconductor spectrometer MDU-Liulin [4]*

The Mobile Dosimetry Unit (MDU) is designed as a handy spectrometer-dosimeter for continuous simultaneous monitoring of the doses and numbers of energy deposition events in a semiconductor detector. It consists of the detector itself, a charge-sensitive preamplifier, 2 microcontrollers, a flash memory and Li-ion cells. Pulse analysis technique is used to measure the deposited energy in the detector. The dose  $D(\text{Si})$  [Gy] is calculated from the spectrum as:

$$D(\text{Si}) = K \cdot \sum (E_{i(\text{dep})} \cdot A_i) / \text{MD}, \quad (1)$$

where MD is the mass of the detector in [kg];  $E_{i(\text{dep})}$  is the energy deposited in the channel  $i$ ;  $A_i$  is the number of impulsion's in it; and K is a coefficient.

The operational time of the MDU depends on the lifetime of the accumulators and on the rate of the memory fills up. In the case of continuous operation the lifetime is 120 hours with standard 1350 mAh Li-ion accumulators, more than 1400 hours with 14 Ah Li-batteries. MDU units were exposed in some on-Earth reference radiation fields. It was found that while  $E_{\text{dep}}$  at photons beams is below 1 MeV, for neutrons it reaches up to 20 MeV; relative

response to neutrons in terms of tissue kerma increases from  $\sim 0.036$  ( $^{252}\text{Cf}$ ) to  $\sim 0.42$  (14 MeV neutrons); at CERF fields  $D(\text{Si}) \sim 0,14 \text{ H}^*(10)$  in high LET part;

### Exposures onboard a CSA aircraft

In the frame of collaboration between NPI AS CR and Czech Airlines it was possible to install a MDU unit inboard of an airbus A310-300. Since the beginning of 2001, six long-term monitoring runs were revised, each close to 2 months long. Description of these runs is given in Table 1. Additionally to that, the measurements were performed during about 30 returned single flights. Examples of results will be also presented in next chapter. Measured data are stored in the flash memory of unit, after the end of exposure they can be transferred to a computer.

Table 1: Long-term onboard monitoring runs realized in 2001-2002

Period	Returned flights monitored <sup>*)</sup>	Numbers of flights
22/03-07/05/01	PRG-NY(25), PRG-TOR(13), PRG-DUB(3)	108
30/05-24/07/01	PRG-NY(41), PRG-TOR(12)	125
29/08-16/10/01	PRG-NY(26), PRG-TOR(13), PRG-DUB(2)	96
25/10-10/12/01	PRG-NY(20), PRG-TOR(7), PRG-AMS(1)	70
06/05-28/06/02	PRG-NY(22), PRG-TOR(13), PRG-DUB(8), PRG-LHR(5), PRG-MAD(5)	124
16/10-06/12/02	PRG-NY(23), PRG-TOR(10), PRG-DUB(1), PRG-DUB-CM(4), PRG-MUN(1)	110

<sup>\*)</sup> PRG – Prague, NY – New York, TOR – Toronto, DUB – Dubai, CMB – Colombo, MUN – Munchen, AMS – Amsterdam, LHR – London, MAD – Madrid

### Treatment of data, examples of results

Treatment of data directly measured with Si-diode energy deposition spectrometer is based on the difference, already mentioned, between energy deposition spectra from photons and other low LET radiations and those from CERF, and onboard exposure. CERF field reference values of  $\text{H}^*(10)$  for high LET component [5], resp. author's values for low LET component are compared with  $D(\text{Si})$  below and above 1 MeV. Also, comparisons of MDU data treated (item 2) and data measured with TEPC were considered [6].

The data obtained this way were always compared with exposure level calculated by means of transport codes CARI 6 [7] and EPCARD 3.2 [8]. One example of comparison is given in the Figure 1 for the flight during quiet solar conditions. One can see there that the agreement of experimental data obtained by means of the interpretation procedure. It can be also seen in Table 2, where the differences between measured values and values as calculated by EPCARD 3.2 are presented for 5 long-term monitoring runs.

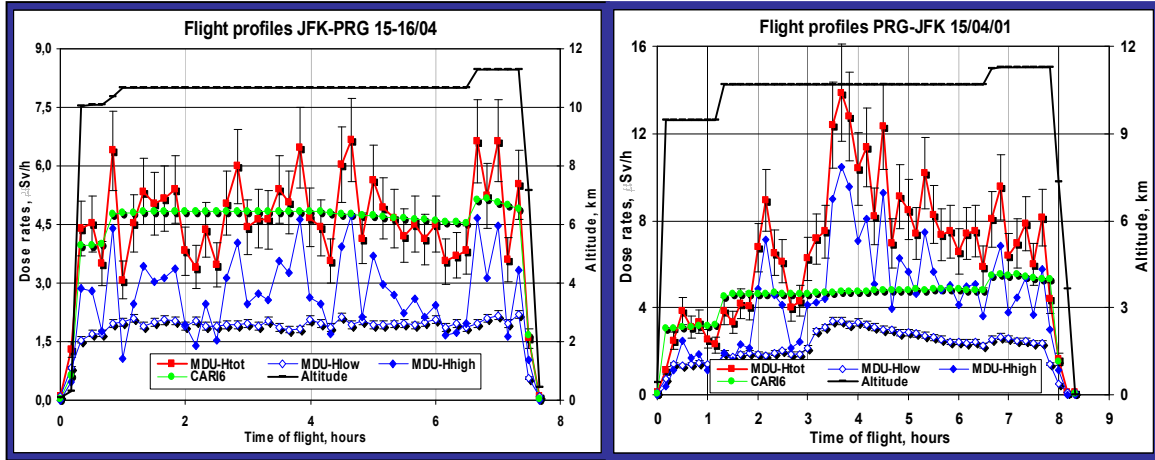


Fig.1: Flight profiles –quiet sun

Fig.2: Flight profile – solar flare

Table 2: Relative deviations for individual flights, in % of total  $H^*(10)$ , of EPCARD calculated and MDU data interpreted as mentioned.

Route	Relative deviation, %, for the flight period				
	22/03-07/05/01	30/05-24/07/01	29/08-16/10/01	25/10-10/12/01	16/10-06/12/02
PRG-JFK	(0.1 5.3)	-(11.5 4.8)	-(5.7 3.7)	-(6.6 5.1)	(4.6 3.5)
JFK-PRG	(0.5 6.0)	-(11.2 5.9)	-(5.6 4.0)	-(7.1 4.6)	(2.9 3.6)
PRG-YYZ	-1.8	-(9.1 1.5)	-(3.7 2.9)	-	-
YYZ-PRG	9.8	-(8.9 1.0)	-(3.1 3.3)	-	-
PRG-YUL	(2.6 6.9)	-(8.9 5.2)	-(5.2 3.6)	-(5.9 4.8)	-(0.4 5.3)
YUL-PRG	(0.6 3.8)	-(10.1 3.5)	-(0.5 5.0)	-(2.5 4.0)	-(0.8 4.8)
PRG-DXB	-(6.7 2.8)	-	-(13.7 0.1)	-	(9.2 3.3)
DXB-PRG	-(8.8 6.4)	-	-(6.8 2.7)	-	(13.4 4.2)

Additionally we present other events registered during our measurements. First, we succeeded to perform measurements during a very rare event, intense solar flare on the 15<sup>th</sup> April 2001, Ground Level Event (GLE) 60. Presence of abrupt increase of exposure level was seen in the full record, the experimental data are compared with the calculation in Fig. 2. One can see there that, at about 3 hours after taking off, the measured dose rate increases abruptly about 3 times. Calculated values rest constant, they consider only galactic not solar component of cosmic radiation. We deduced, that the total ambient dose equivalent increased by about 40%, comparing to only galactic component.

Other examples are presented in Figures 3 and 4, where experimental and calculated data are compared for a flight over the equator, from Paris to Brisbane via Singapore. One can see there important influence of geomagnetic position. When the aircraft is flying from Paris to Singapore, the exposure level decreases for the same altitude with increasing geomagnetic rigidity (~4 GV at Paris, ~17 GV at Singapore). Opposite effect is seen for the flight from Singapore to Brisbane when geomagnetic rigidity increases again (~8 GV).

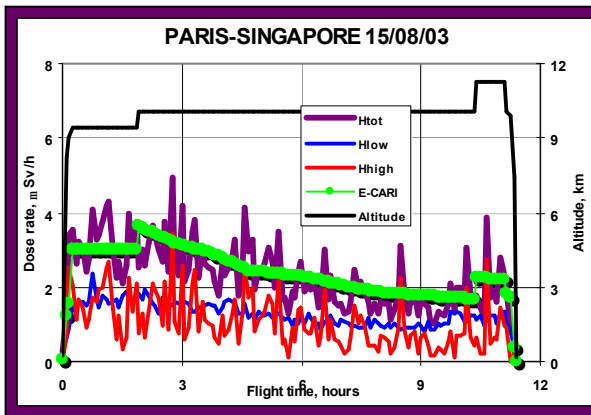


Fig. 3: Flight profile – route to equator

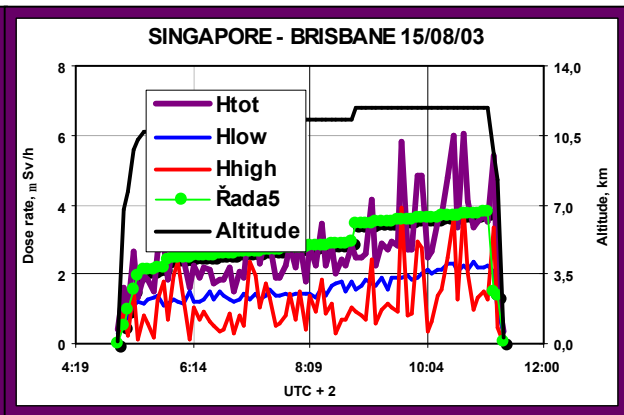


Fig.4: Flight profile – route to south pole

## Conclusions

The semiconductor spectrometer proved the possibility to monitor dosimetric characteristics of radiation fields on aircraft board during about 2 months. It permitted to register onboard a ground level solar event. The results obtained can significantly help to estimate the contribution of solar eruptions to aircrew exposure.

Spectrometric properties of it permit to enlarge the interpretation of directly registered data. An additional effort is needed to improve this performance. Further calibration in the CERF fields is needed, important would be to acquire further onboard data in cases when the contribution of the ionizing/neutron part of the field changes.

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# **Radon/Thoron Measurements of Working Campus Building of Istanbul University**

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## **ABSTRACT**

This study has been carried out in order to determine radon, and radium concentrations in air, in soil and in the construction. The materials of old wooden houses and of the concrete buildings surrounding old part of Istanbul University campus. The location comprising buildings of different ages and structural compositions. Concentration levels of  $U^{238}$ ,  $Th^{232}$ ,  $K^{40}$  and  $Cs^{137}$  radionuclide shows significant variation from one building to the next. The CR-39 type plastic track detectors and track etching technique was used in measuring the concentrations. Data from about 100 units shows that indoor radon and thoron concentration vary from 10.3 Bq/m<sup>3</sup> to 190 Bq/m<sup>3</sup> and 1 Bq/m<sup>3</sup> to 144 Bq/m<sup>3</sup> respectively. In general, the level of radon was observed highest in winter and lowest in summer times. Detailed analyses of radon and thoron distribution for all units with seasonal variations are presented in the study.

## **I. INTRODUCTION**

The inhalation of  $^{222}Rn$  decay products in living environment results in large fraction of the total human exposure to ionizing radiation. Statistics show that indoors  $^{222}Rn$  exposure may be responsible for more than 10 % incident of lung cancer in some countries.  $^{222}Rn$  naturally occurring element is present in trace amount in the earth's crust and decays with a 3.8 days half- life into a short lived particles and attaches themselves to aerosol particles. Inhalation and retention of these highly energetic particles in lung can elevate the cancer risk.  $^{222}Rn$  and its decay products rarely reach elevated levels outdoors due to continual air dispersion and dilution. Indoors, due to poor ventilation in the buildings may enable Radon and its products to reach high level concentrations with respect to outdoor level. On the other hand thoron concentration in dwellings is considered negligible because of it's the short half-life time (55.6 sec).

While ionizing Radon-222 and its products their possible health effects have been studied widely, large-scale survey to determine frequency distribution of  $^{222}Rn$  exposure also concerned scientists is therefore important to know the frequency distribution of  $^{222}Rn$  in human environment. Keeping this fact in view, we designed a survey taking into consideration that its main goal is to obtain the annual average in door radon concentration in different parts of geographical region of our interest. In the spring 1998, the research project was started as a master thesis project by on of us (Mrs. Sezen). A twin dosimeter was used for simultaneous measurements of radon and thoron inside the dwellings. About 100 houses are surveyed in the main campus of Istanbul University.

## II. MATERIALS AND METHODS

### II.1. Gamma Rays Spectrometer (Canberra-90)

One of the objectives of this study was to identify and determine natural radionuclides activity concentrations in the designed area for that reason two types of soil samples were collected at 0-10 cm depth level from the uncultivated fields. The collected samples were weighed individually, dried, pulverized, homogenised and sieved to pass through 1 mm mesh. Meshed soil samples were transferred to Marinelli beakers, which have 1000 ml capacity. Subsequently and also the samples are weighed and carefully sealed for 30 days to reach secular equilibrium for radium, radon and their respective progeny. Further the process each soil samples were counted including the background, which had 50000 seconds typical counting time...

To evaluate the concentrations of radionuclides, the detector efficiency as a function of energy for the same counting geometry was considered. The concentrations of different radionuclides were calculated based on the measured detector efficiency. Radionuclides which are present in situ soil samples are  $^{226}\text{Ra}$ ,  $^{214}\text{Pb}$  and  $^{214}\text{Bi}$  which are from  $^{238}\text{U}$  series and  $^{228}\text{Ac}$ ,  $^{212}\text{Pb}$ ,  $^{212}\text{Bi}$  and  $^{208}\text{Tl}$  which are from  $^{232}\text{Th}$  series  $^{40}\text{K}$  and  $^{137}\text{Cs}$  respectively.

### II.2. Radon Detectors

The radon measuring probes used in this investigation were the passive nuclear track detectors made of by 250-micrometer thick CR-39 foils. The CR-39 foils were placed in 10 cm<sup>3</sup> polystyrene cups plastered with polystyrene cap through which radon may pass by diffusion while radon progeny and aerosols are retained on the surface of the cap. A tree of the detector cups was distributed to old wooden houses and campus buildings. Two cups was placed in each floor of the premises including the basements. The detectors were put one meter above in the each floor and left for about three months before replaced with the second set of them and the procedure were repeated for 2 terms. In addition, several detectors were kept in sealed aluminum foils for background evaluation during the measurements. Collected detector foils were chemically etched in 30% NaOH solution at 70 °C for 17 hours. The tracks were counted under an optical microscope. The calibration factors for old wooden houses and campus buildings were determined separately for each set and verified analytically. Background tracks were subtracted from the tracks of exposed detectors and average concentration values were evaluated. Calibration measurements were made using calibration chamber of 222-L oil drum containing  $^{226}\text{Ra}$  source, maintaining a concentration of about 3.2 kBq.m<sup>-3</sup> at TAEK (Turkish National Atomic Agency) Laboratories.

## III. RESULTS AND DISCUSSION

The collected data from old wooden houses are given in Fig...3.1 and Fig... 3.2. The concentrations of indoor radon are found to vary from a minimum value of 10 Bq m<sup>-3</sup> to maximum value of 161 Bq.m<sup>-3</sup> respectively. For the campus buildings indoor radon concentration varies from a minimum value of 55 Bq.m<sup>-3</sup> to a maximum value of 483 Bq.m<sup>-3</sup>.

Most of the houses in the campus area are constructed from woods, local stone and bricks, whereas the buildings were constructed from cement, stone and mortar. In general the radon concentration were found higher in concrete buildings than that of old wooden houses,

which is also justify our investigation purposes, where basement floors of such houses are constructed on poured concrete. The basement floor allows more radon to diffuse inside the building because of higher porosity of material used. The emanation of radon is also higher from rocks and ground floor stones. In addition, basement dwellings have small doors and window (some of them are without window), which remains closed for most of the time to conserve energy in winter time. Due to poor ventilation conditions, radon is accumulated inside the premises and thus results in higher concentration of activity. The result is given Fig... 3.3 and Fig...3.4.

In this study the construction material were taken from old wooden houses shows <sup>226</sup>Ra concentration of 15.86 Bq/kg for the basement and 67.4408 Bq/kg for the brick, which were used in the construction material for the fire walls. The fire wall bricks resulted highest concentration, when it compared to other construction materials used. As of the consequences radionuclei concentration from firewalls bricks for <sup>226</sup>Ra, <sup>214</sup>Pb, <sup>232</sup>Th, <sup>228</sup>Ac gives high degree of doses, which can be seen. in Fig. 3.6.

The in door radon concentrations in old wooden houses were examined for two terms; the first period between from (03 08 1998 – 03 04 1998) gives arithmetic mean value of 98 Bq/m<sup>3</sup> and geometric mean value of 93 Bq/m<sup>3</sup>. The second period ,which is between (03.04.1998-03.07.1998) gives the arithmetic mean value of 47 Bq/m<sup>3</sup> and geometric mean value of 38 Be/

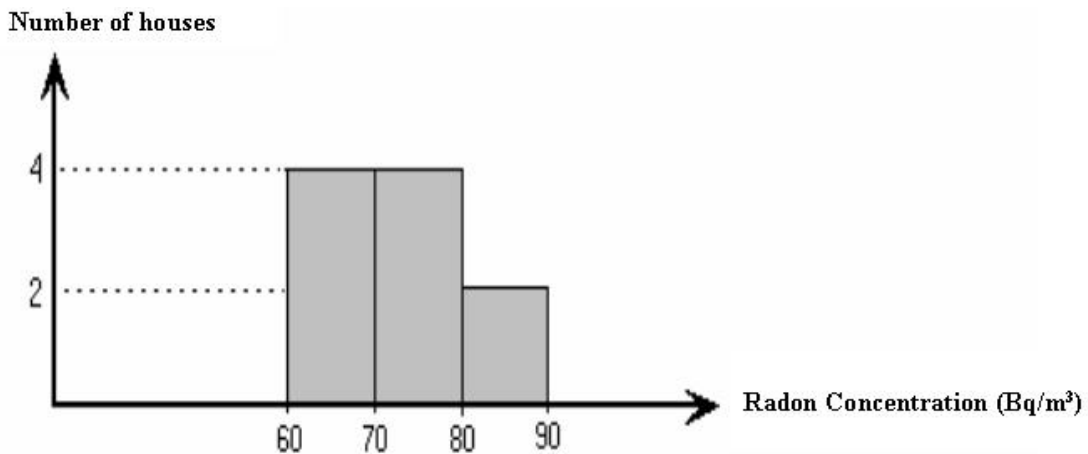


Figure 3.5- Overall radon concentration of old wooden houses



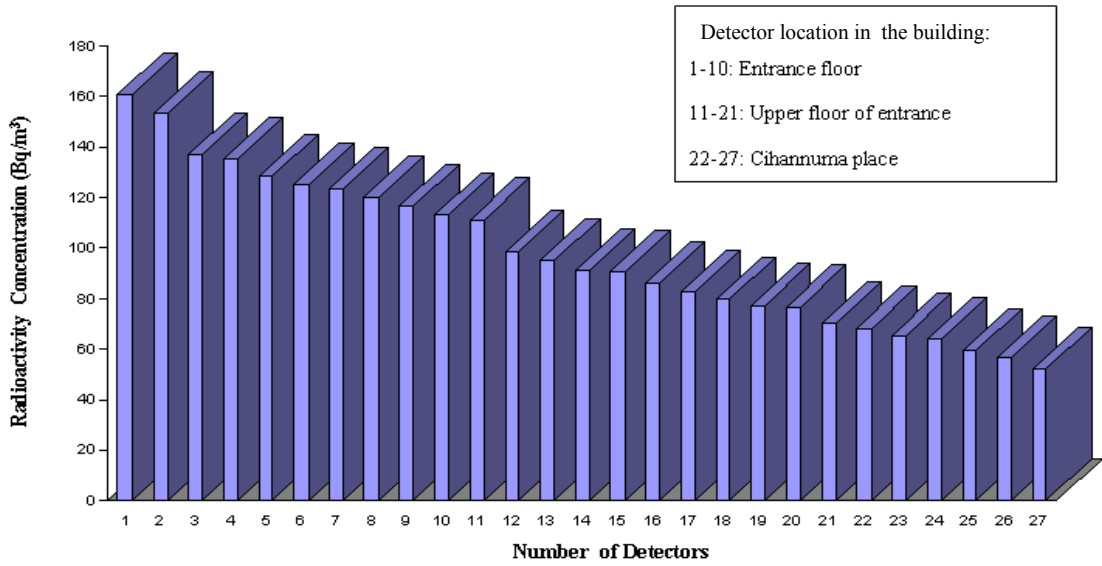


Figure 3.1- Indoor radon concentrations obtained from old wooden houses between dates 3.1.1998 – 3.4.1998

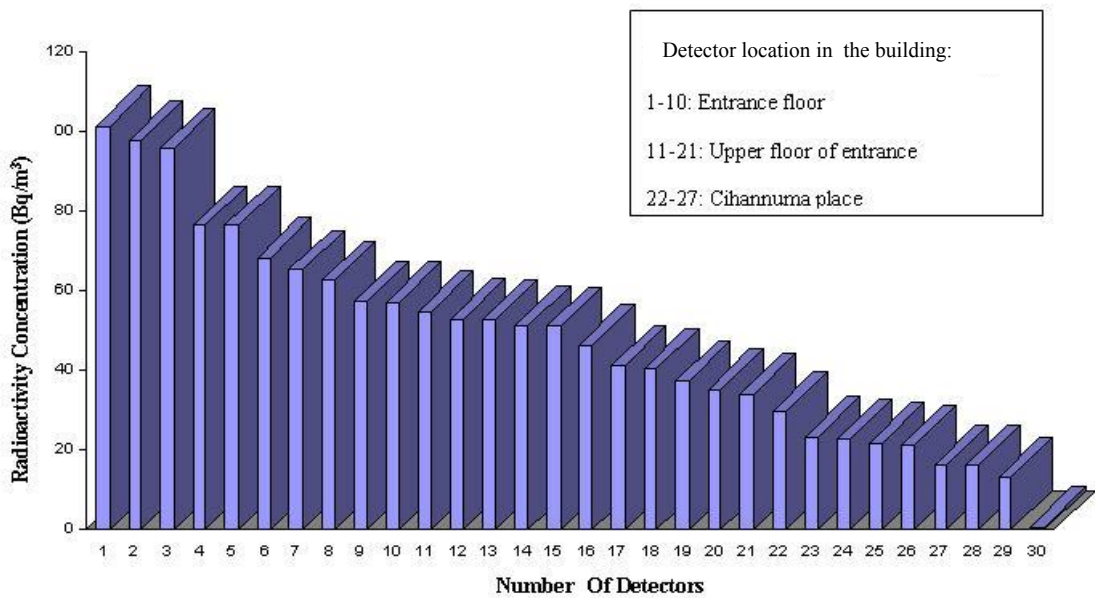


Figure 3.2- Indoor radon concentrations obtained from old wooden houses for the period 5.4.1998 – 3.7.1998

According to UNCEAR's 2000 reports suggest that the indoor radon concentration, for the world general public, should be  $40 \text{ Bq/m}^3$  for the arithmetic mean and  $30 \text{ Bq/m}^3$  for the geometric mean values. The arithmetic mean value in wooden houses for both terms is found to be  $73 \text{ Bq/m}^3$ , which is almost twice the world arithmetic overall. In old wooden houses, overall measured value is  $50 \text{ Bq/m}^3$ . One of the reasons of high radon concentrations in old wooden houses is due to the high radium in construction materials. In addition, further modifications done by householders and the cracks in the basement floor is the caused can be other reason.

As can be seen from the figure 3.1 and Figure 3.2\ the radon concentration values of old wooden houses for the first and second terms show variant values as expected.

The measured values for the second terms are much lower and the value decreases with increasing floor levels. The minima and maxima for this period are 63 Bq/m<sup>3</sup> and 89 Bq/m<sup>3</sup> respectively.

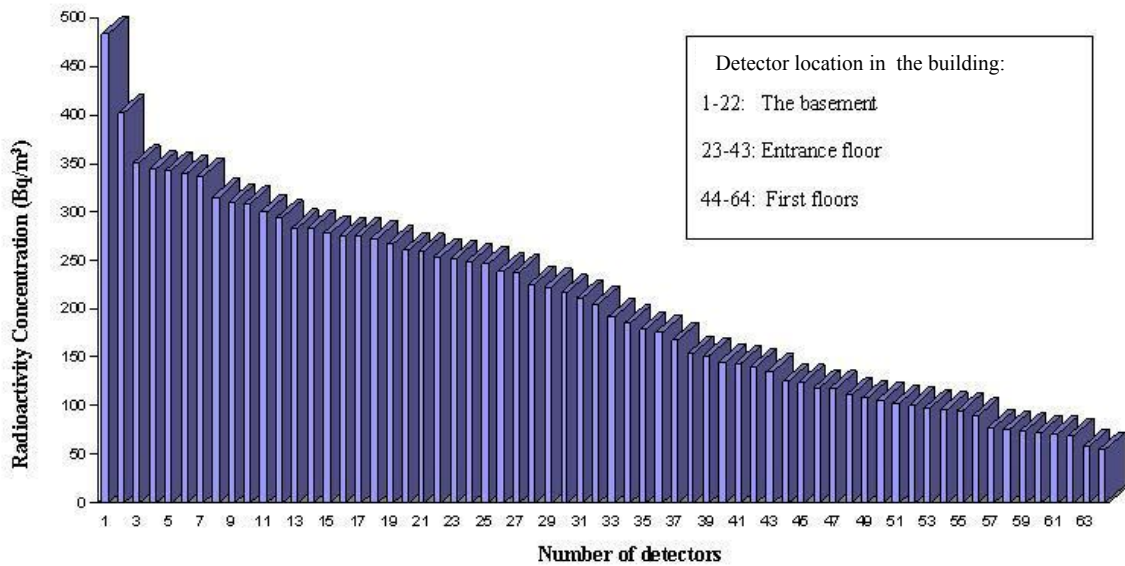


Fig.ure 3.3- Indoor radon concentrations for 3 campus buildings for the period 3.11.1999 – 3.1.2000

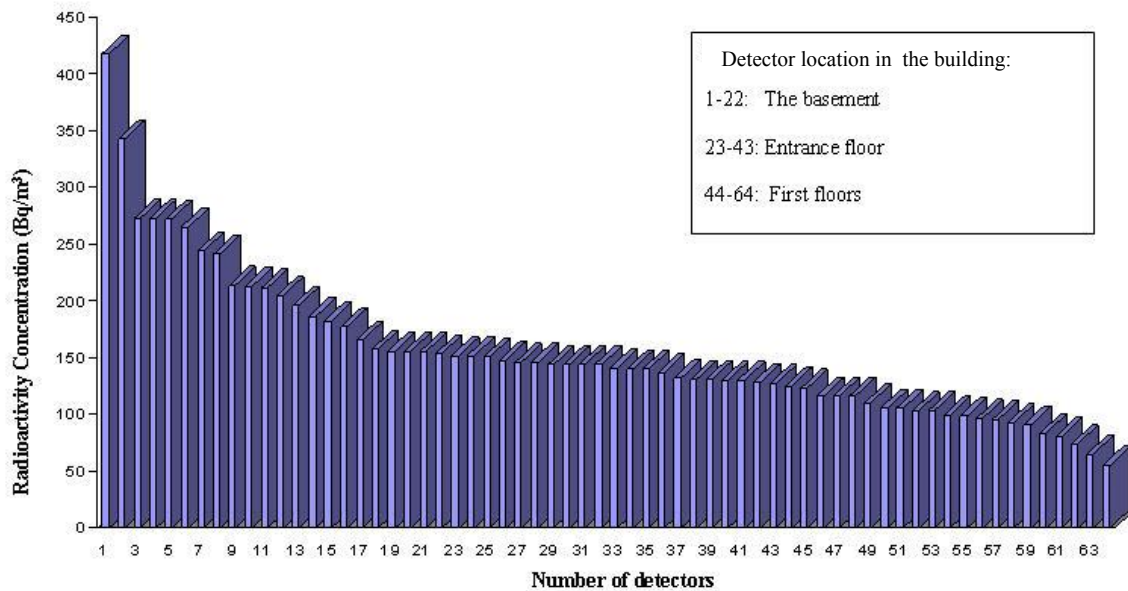


Fig.ure 3.4-Radon concentrations in three departments for the period 3.11.2000 – 3.3.2000

For in door radon concentration ICRP-65 reports recommends for houses is 200-600 Bq/m<sup>3</sup> and for the Workstation is 500-1500 Bq/m<sup>3</sup>. Turkish National Atomic Agency ( TEAK) also recommends for the Turkish public 400 Bq/m<sup>3</sup> for the house hold and 1000 Bq/m<sup>3</sup> per year for the working places.

Table 3.1- Detector placement via buildings

<b>DETECTOR PLACEMENT VIA BUILDINGS</b>	
<b>Physics department</b>	<b>1, 3, 6, 9, 12, 15, 18, 21, 24, 27, 30, 33, 36, 39, 42, 45, 48, 51,54, 57,60, 63</b>
<b>Biology department</b>	<b>2, 4, 7, 10, 13, 16, 19, 22, 25, 28, 31, 34, 37, 40, 43, 46, 49, 52, 55, 58, 61, 64</b>
<b>Student cultural centre</b>	<b>5, 8, 11, 14, 17, 20, 23, 26, 29, 32, 35, 38, 41, 44, 47, 50, 53, 56, 59, 62</b>

The results for the physics department building, which is constructed in natural stone, are given in Fig. 3.6. The values for  $^{226}\text{Ra}$  and  $^{232}\text{Th}$  concentration are 13.513 Bq/kg and 5.7 Bq/kg respectively. Indoor radon concentration for the department was conducted for two consecutive terms. First term evaluation was carried out between (3.11.1999 -3.1.2000) and second one between (3.1.2000-3.3.2000). During these two terms, the ventilation was low due to winter conditions. At basement floor, radon concentrations were found higher; however these parts were separate and are used rarely.

Computed arithmetic mean and geometric mean values for two terms are for first term are 208.54 Bq/m<sup>3</sup>, 179.83 Bq/m<sup>3</sup> and for second term are 160.77 Bq/m<sup>3</sup>, 146.90 Bq/m<sup>3</sup> respectively. The present values are well under TAEKs tolerable regime. The concentrations of radon were even lower for the upper parts of the building. As can be seen from the fig. 3.3 and fig.3.4.

Gamma spectroscopic results for the wall construction materials, as well as the wall plaster for the biology department are given in Fig... 3.6. From the figure, one can see that the  $^{226}\text{Ra}$  concentration is 30 .69 Bq/kg and  $^{232}\text{Th}$  concentration is 16.62 Bq/kg. For the department, indoor radon also were performed in the similar fashion first term was between (3.11.1999-3.1.2000). The arithmetic mean and geometric mean values for this term are 200.63 Bq/m<sup>3</sup> and 174.10 Bq/m<sup>3</sup> respectively. Arithmetic and geometric mean value for second term for the period (3.1.2000- 3.3.2000) are 153.77 Bq/m<sup>3</sup> and 141.65 Bq/m<sup>3</sup> respectively. Again these results were under TAEK's tolerable limits and the results are given in Fig... 3.3 and Fig... 3.4. From the figures, the concentration of radon was even lower for the upper parts of the building. One of the soil sample was taken from biology department garden and the analysis for garden soil were had similar chemical composition as other part of soil in Istanbul and the results are shown in figure.3.7

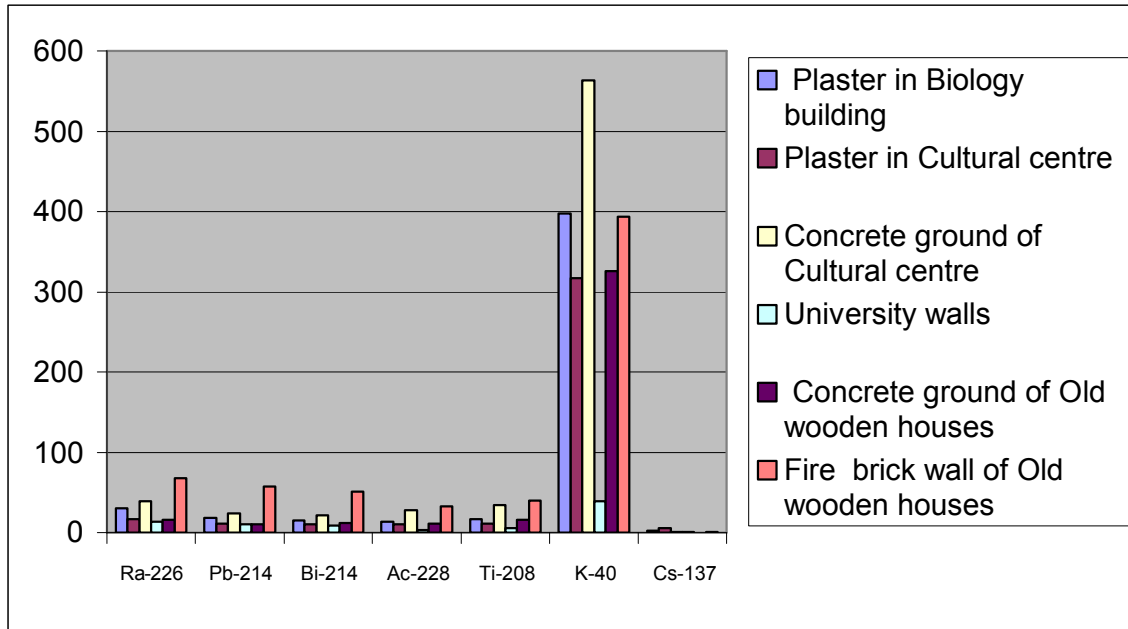


Figure 3.6- Radioactivity concentrations of  $^{238}\text{U}$  series,  $^{232}\text{Th}$  series,  $^{40}\text{K}$  and  $^{137}\text{Cs}$  in construction material samples

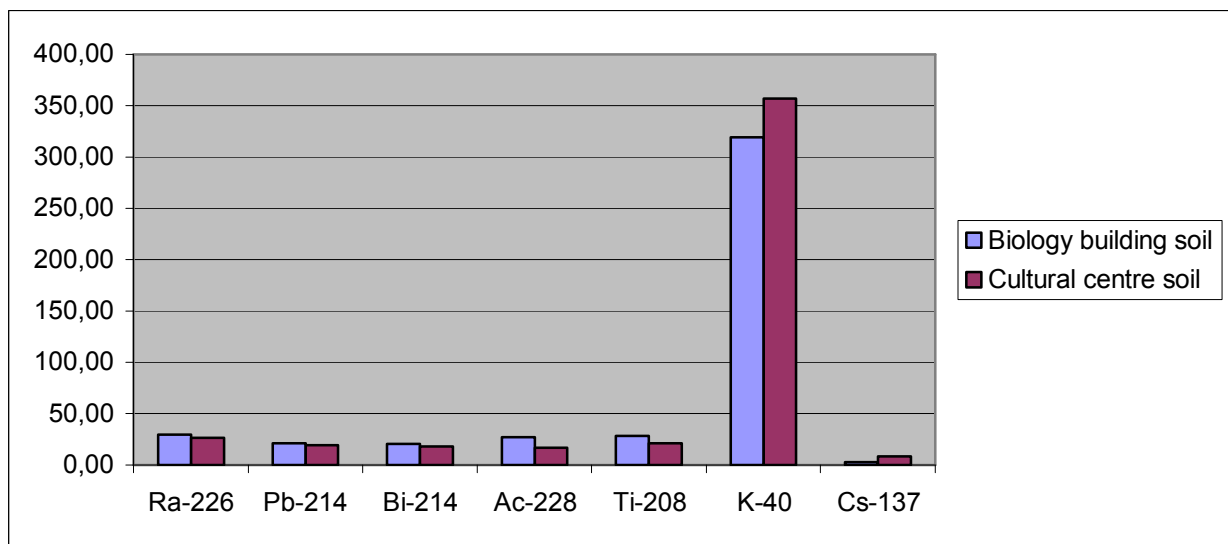


Figure 3.7- Radioactivity concentrations of  $^{238}\text{U}$  series,  $^{232}\text{Th}$  series,  $^{40}\text{K}$  and  $^{137}\text{Cs}$  in soil samples

Similarly, spectroscopic analyses were conducted using the samples taken from inner part of the wall plasters and ground concrete floor of the student cultural the results are given in Fig. 3.6. For the ground concrete floor  $^{226}\text{Ra}$  and  $^{232}\text{Th}$  concentrations are 38.80 Bq/kg, 34.12 Bq/kg and for wall plaster radionuclide concentration is 16.35 Bq/kg, 11.44 Bq/kg.

The highest concentration for  $^{40}\text{K}$  was 563.48 Bq/kg as shown in figure 3.6. Even the building itself has no basement floor, the values were obtained is too high, one reason attributed to the cause is that, the building does not have proper ventilation system and windows are always shut close. In the rooms and winter conditions. Again, the concentration is still under TAEK's tolerable limits. Indoor radon concentration was done as again for two

terms. Arithmetic mean and geometric mean values for first term between (3.11.1999-3.1.2000) are 192.6 Bq/m<sup>3</sup> and 171.78 Bq/m<sup>3</sup> respectively. For the second term indoor radon concentrations arithmetic mean and geometric mean values are 146.5 Bq/m<sup>3</sup> and 139.16 Bq/m<sup>3</sup> respectively.

#### **IV. Conclusion:**

Indoor radon concentration analyses and the mean values are still under TAEK's tolerable limits for the campus. These results show us that both householders and university people weren't affected from radon doses gathered by inhalation per a period of year.

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# TEN YEARS OF CONTINUAL MONITORING OF $^{222}\text{Rn}$ CONCENTRATION IN BRATISLAVA ATMOSPHERE

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## **Introduction**

The radon ( $^{222}\text{Rn}$ ) activity concentration in the surface layer of the atmosphere is not stable. It depends mainly on meteorological parameters as the solar radiation, the wind velocity and the cloudiness. However, there were found regular daily and seasonal variations of the  $^{222}\text{Rn}$  concentration in the surface layer of the atmosphere. These variations are explicitly attributed to the regular changes of the rate of the vertical air mixing [1].

In 1983 Gessell [2] gave a review of some basic papers pertinent to the radon variations. However, the research works paying an attention to  $^{222}\text{Rn}$  in the outdoor atmosphere are topical even after many years. In the last ten years the motivation for the radon investigation in the lower atmosphere has been the possibility of the utilization of  $^{222}\text{Rn}$  as a tracer in many atmospheric studies [3 - 6]. The study of  $^{222}\text{Rn}$  behaviour in the lower atmosphere is also important for more precise estimation of the public radiation dose due to inhalation of  $^{222}\text{Rn}$  and its decay products. The developing of new radon models contributes to the better understanding of the radon behaviour in the surface layer of the atmosphere under variable meteorological conditions and to the prediction of radon concentration from the meteorological data.

The atmospheric radon concentrations can vary substantially with location. Therefore radon measurements on the regional level bring almost always new knowledge concerning the

behaviour of the  $^{222}\text{Rn}$  in the lower atmosphere and the population exposures. In our contribution the results of the long – term measurements of the radon concentration in Bratislava atmosphere are presented and complex view on daily and seasonal radon variations is provided.

## Methods

The measurements of radon were carried out at the open grass area in the campus of the Faculty of Mathematics, Physics and Informatics in Bratislava (48° 9' N, 17° 7' E, 164 m a. s. l.). The air for the analysis was sucked at a height of 1.5 m above the ground surface. The radon activity was continuously monitored using the large volume scintillation chamber [7]. The flow rate of the air through the chamber was selected  $0.5 \text{ l}\cdot\text{min}^{-1}$  in order that  $^{220}\text{Rn}$  to be decayed still before the inlet of air into the chamber. The sensitivity of the scintillation chamber is 0.3 cpm at  $1 \text{ Bq}\cdot\text{m}^{-3}$  of the  $^{222}\text{Rn}$  activity concentration in the entering air. The radon monitor allows obtaining of almost 80 % of the data of the radon activity concentration in the surface layer of the atmosphere with uncertainties less than 30% in the counting interval of 2 hours.

## Results and discussion

$^{222}\text{Rn}$  in the surface layer of the atmosphere in Bratislava was measured from 1987 on and since 1991 it has been monitored continuously with frequency of 12 data of  $^{222}\text{Rn}$  activity

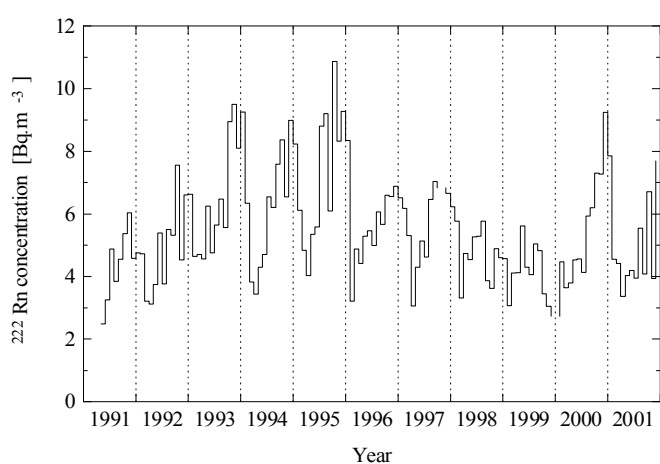


Fig. 1. The monthly mean values of the  $^{222}\text{Rn}$  activity concentration in the surface layer of the atmosphere in Bratislava.

concentration per day. In this work mainly the results of the measurements in the period 1991 – 2000 are presented. The extensive set of approximately 39 000 data obtained by continual monitoring allows to study the average daily courses of the  $^{222}\text{Rn}$  concentration for individual months and different years and to demonstrate the average annual courses even in

different time period of the day. In addition it enables us to map long – term trends of the  $^{222}\text{Rn}$  activity concentration in the atmosphere.



The monthly means of the radon activity concentration for the years 1991 – 2001 are shown in the Fig. 1. The spring minima and maxima occurring in various months of the second half – year can be seen in the annual  $^{222}\text{Rn}$  courses for all the years. However, the individual years differ from each other quite considerably. The amplitudes of the annual courses vary from  $3.1 \text{ Bq.m}^{-3}$  (1995) to  $1.2 \text{ Bq.m}^{-3}$  (1999). The years 1996 – 1999 show a decreasing trend of these amplitudes, predominantly as a consequence of the decreasing of the radon activity in the year's maxima. Simultaneously, a shift of the year's maxima from the late autumn and winter months towards the summer months is observed. This effect is due to the lowering of the radon activity concentrations in the minima of the daily cycles of the autumn and winter months with only mild change of amplitudes of the daily  $^{222}\text{Rn}$  waves. This behaviour of the daily  $^{222}\text{Rn}$  waves is clearly seen in Fig. 2 where the monthly mean diurnal waves of the  $^{222}\text{Rn}$  activity concentration are compared for the years 1994 and 1998.

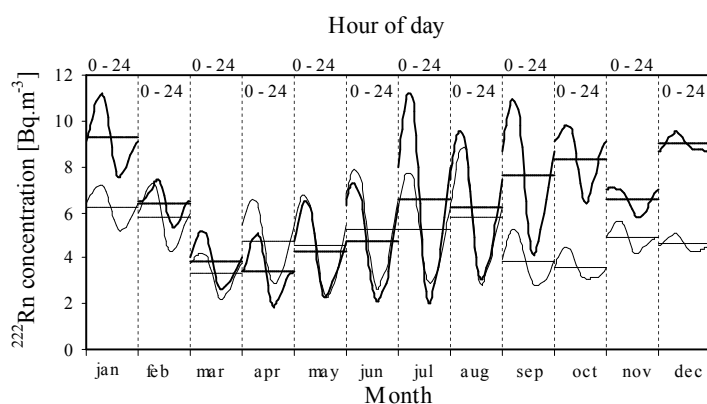


Fig. 2. The monthly mean diurnal  $^{222}\text{Rn}$  waves in Bratislava atmosphere for the years 1994 (thick curve) and 1998 (thin curve).

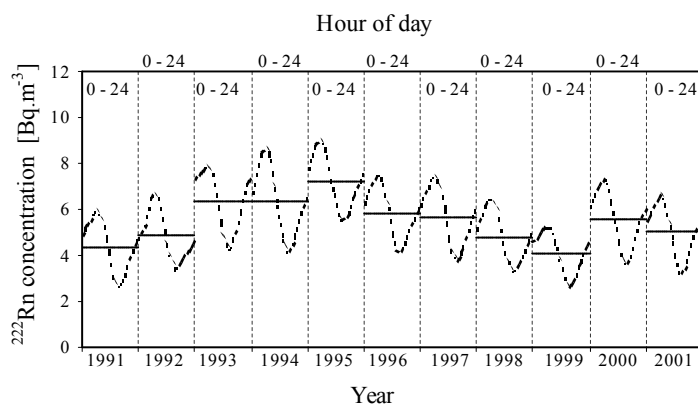


Fig. 3. The mean annual values (solid lines) and the mean diurnal courses (dotted curves) of the  $^{222}\text{Rn}$  activity concentration in the Bratislava atmosphere.

The observed changes of the annual and daily radon courses can indicate the lowering of the stability of the surface layer of the atmosphere in autumn and winter months in the last period in comparison to the previous years.

The long – term trend of the average annual  $^{222}\text{Rn}$  activity concentrations is shown in Fig. 3. For individual years, there are presented also average radon daily courses.

The average annual radon activity concentrations vary from  $4.1 \text{ Bq.m}^{-3}$  (in 1999) to  $7.2 \text{ Bq.m}^{-3}$  (in 1995).

The average radon activity

concentration in years 1991 – 2000 is equal to  $5.6 \text{ Bq.m}^{-3}$ . The highest radon concentration in 1995 is caused by the high amplitude of average daily wave in this year ( $1.7 \text{ Bq.m}^{-3}$ ) and also by high average value of the radon concentration during a day ( $5.6 \text{ Bq.m}^{-3}$ ). In 1999 the minimum of the average daily wave is equal  $2.6 \text{ Bq.m}^{-3}$  and its amplitude is equal only to  $1.2 \text{ Bq.m}^{-3}$ . In investigated period, the highest amplitude of average daily wave was observed in 1994 ( $2.3 \text{ Bq.m}^{-3}$ ).

The average annual course of the  $^{222}\text{Rn}$  activity concentration calculated on the basis of all the continual measurements in years 1991 – 2000 is shown in Fig. 4. This course reaches the maximum in months from October to January ( $6.9 \text{ Bq.m}^{-3}$ ) and the minimum in April ( $3.9 \text{ Bq.m}^{-3}$ ). The variation coefficient  $V_{A(\text{Rn})}$  (the square root of the variance divided by the mean) of average annual course is equal to 0.2 and it varies for the radon annual courses in the period 1991 – 2000 from 0.18 (1998) to 0.30 (1994).

In the Fig.4 there are also shown the average annual radon courses for different time periods of the day. The annual courses vary significantly during various hours of the day. For early morning hours the annual courses present maxima shifted to the summer months (August), meanwhile during daily hours (between 10 a. m. – 8 p.m.) the annual courses of the  $^{222}\text{Rn}$  activity concentration have the maxima shifted to the winter months and minima are extended in several spring and summer months (April – July).

The variability of the annual courses of the  $^{222}\text{Rn}$  activity concentration in twelve different time periods of the day is shown in Fig. 5. The variation coefficients and also amplitudes (the amplitude  $A \sim 1,4 V_{A(\text{Rn})} A_{\text{Rn,m}}$ , where  $A_{\text{Rn,m}}$  is the mean annual radon activity concentration) of the annual courses of the  $^{222}\text{Rn}$  activity concentration are low and almost constant for night and early morning hours ( $V_{A(\text{Rn})} \sim 0,17$ ) and they reach maxima during the time interval 2 p. m. – 4 p. m. ( $V_{A(\text{Rn})} \sim 0.37$ ). By means of these results it can be explained why the published annual courses of the  $^{222}\text{Rn}$  activity concentration measured in various periods of the day differ from each other although the measurements were performed in mutually close localities [2].

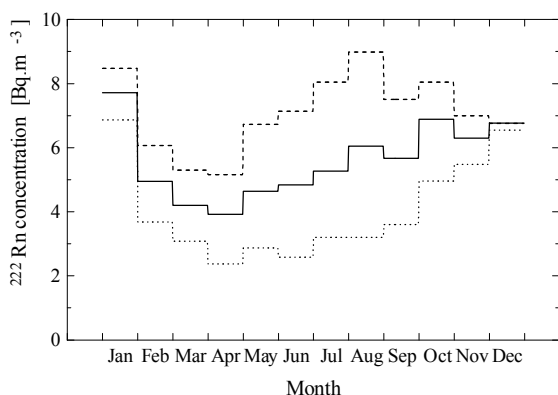


Fig. 4. The mean annual courses of the  $^{222}\text{Rn}$  activity concentration in Bratislava atmosphere for years 1991-2000 (— mean of all values, ..... mean in time period 4 - 6 a.m., --- mean in time period 2 - 4 p.m.).

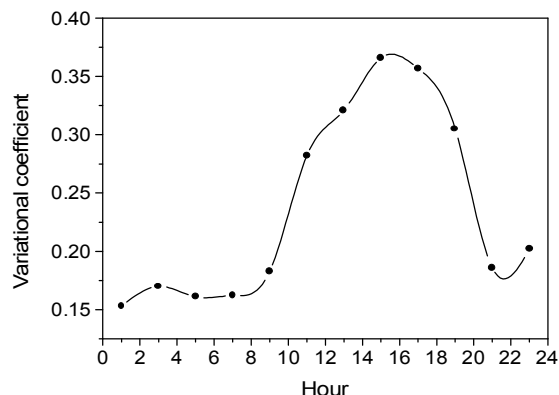


Fig. 5. The relative variability of the mean annual courses of the  $^{222}\text{Rn}$  activity concentration for the different time periods of the day (years 1991 - 2000).

The average daily wave of the  $^{222}\text{Rn}$  activity concentration and waves for some months obtained as the mean of all data from years 1991 – 2000 are shown in Fig. 6. We can see that the average  $^{222}\text{Rn}$  activity concentration reaches a maximum between 4 and 6 a. m. ( $7.1 \text{ Bq.m}^{-3}$ ) and a minimum between 2 and 4 p. m. ( $4 \text{ Bq.m}^{-3}$ ). The  $^{222}\text{Rn}$  activity concentration reaches its average daily value equal to  $5.6 \text{ Bq.m}^{-3}$  at about 10 a.m. and at 9 p.m.

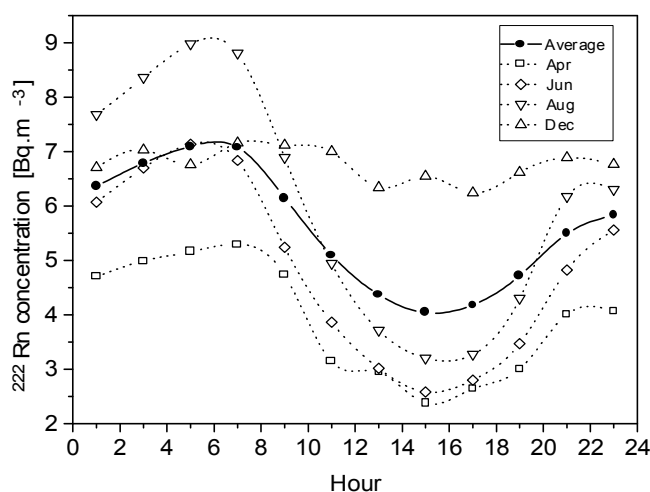


Fig. 6. The mean diurnal courses of the  $^{222}\text{Rn}$  activity concentration in Bratislava atmosphere (years 1991 - 2000).

Also the average daily courses of the  $^{222}\text{Rn}$  activity concentrations for individual months have a form of waves with the maximum in the morning hours and with the minimum in the afternoon. The maximal amplitudes of the daily waves are reached in the summer months from June till August ( $2.2 - 2.9 \text{ Bq.m}^{-3}$ ). In this part of the year also the lowest afternoon  $^{222}\text{Rn}$  activity are reached ( $\sim 3 \text{ Bq.m}^{-3}$ ). The amplitudes of the daily waves are very small at the end of autumn and during winter months ( $0.5 - 0.7 \text{ Bq.m}^{-3}$ ).

In our previous work it was shown that the amplitudes of the daily waves are in proportion to the intensity of the global solar radiation [8].

## Conclusion

By the continual monitoring we obtained the extensive set of radon data in Bratislava atmosphere covering the time period of 1991 – 2000.

The average annual radon activity concentrations varied from 4.1 to 7.2 Bq.m<sup>-3</sup>. In the years 1996 – 1999 the decreasing of the average annual radon concentration was observed.

The average daily courses of the radon activity concentration for individual months calculated on the basis of all data from 1991 – 2000 have a form of waves with the maximum in morning hours and with the minimum in the afternoon. The maximal amplitude of daily wave was found out in August (2.9 Bq.m<sup>-3</sup>) and minimal in December (0.5 Bq.m<sup>-3</sup>). The average daily wave obtained as the mean off all data from years 1991 – 2000 reaches the maximum between 4 and 6 a. m. and the minimum between 2 and 4 p. m.

The <sup>222</sup>Rn activity concentration reaches its average daily value equal to 5.6 Bq.m<sup>-3</sup> at about 10 a. m and at 9 p. m. The amplitude of average daily wave is equal to 1.5 Bq.m<sup>-3</sup>.

The average annual radon course calculated on the basis of all the measured data reaches the minimum in April and the maximum in October with seasonal variation from 3.9 to 6.9 Bq.m<sup>-3</sup>. The annual radon courses differ from each other for various periods of the day.

## Acknowledgements

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# GEOSTATISTICS APPROACH TO RADON POTENTIAL MAPPING

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## **Introduction**

Currently used procedures for the meso-scale radon potential mapping are in general based on the integral evaluation of certain number regularly distributed soil-gas concentration samples with reconnaissance scale ranging from few meters to hundreds, direct interpretation of a geology structure or an airborne radiation mapping. It's generally recognized that the soil radon survey provides an important guide for the assessing radon potential in the buildings, although Reimer (1995), Húlka et al. (1994) and other authors proved striking imbalance between house-measured radon and the results of this survey. Since several studies proved high radon variability over the time and space, even over the small distances (Ort et al. 1995), the integral approach mentioned above could result in the misclassification of the areas with elevated radon potential and in the vagueness of further exploitation. The solution we have used lies in the full integration of spatial aspect into the radon potential mapping, exact analyses of source data structure inclusive all uncertainties, and estimation of complex radon field. This approach is useful for the meso-scale radon potential mapping instead of, or as a complement to standard methods. This also provides precise classification of the areas with elevated soil-radon potential, inclusive uncertainties affecting this process. Similar technique has also been used by Voltz and Webster (1990), and Oliver and Kharyat (1999). In particular, we focused on the analyses of the spatial structure of radon soil-gas concentration samples, on radon field temporal variability evaluation and on the prediction of continuous radon field by means of chosen geostatistics tools. The research has been conducted in the area neighboring to Banská Bystrica city (Central Slovakia, fig. ) approximately 5 km (E-W) 2 km (N-S) of size. The geological background of the area is composed of *Mesozoic carbonate rocks*, which are covered by *Tertiary sediments* and *volcanic rocks*. The higher radon concentration can be preliminary expected due to the andesitic volcanoclasties, but might be complicated by two factors – assumed NE-

SSW and NW-SE fault systems and possible presence of underlying *Permian sedimentary rocks*. To collect radon samples the random sampling scheme has been proposed (fig. 1). Primarily more than 100 sampling points have been designed, but during the field data gathering this number had been reduced due to the ground water and stony areas occurrence. Soil-gas samples have been collected using a method described by Neznal and Pernicka (1996). The data gathered followed lognormal distribution averaging between 46 800 – 61 100 Bq/m<sup>3</sup> (95% confidence interval). Descriptive statistics can be seen in table 1. In addition, two nested sampling schemes have been proposed – the first one consisting of 50 points and the second one consisting of two sampling fields focused on large-scale investigation.

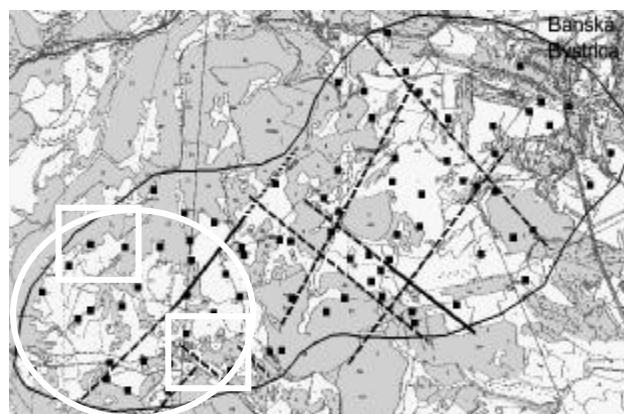


Fig. 1. Research area. Black dots express the primary data field, black solid and dashed lines are proved and expected geology gaps, dark gray are forested areas. White circle delineates the locations where the long term monitoring has been conducted. Squares delineate further specific sampling areas.

Tab. 1 Primary radon field descriptive statistics (Bq/m<sup>3</sup>)

N	Mean	St.dev	Median	Min	Max	25% Q	75%Q	IQR	Range
87	53 970	33 678	47 900	8 198	144 600	26 658	73 525	46 867	136 411

### Spatial Variability Evaluation

Collected point samples and related descriptive statistics provide sufficient information to judge the overall radon potential of the research area. However, with regard to proved high spatial variability of radon activities over the space, the use of such a model could result in an unwanted deviation of the radon risk assessment and further respective analyses. Therefore we also focused on the radon spatial variability evaluation by means of variographic analysis. Individual variograms constructed in different directions proved in general an intensive spatial variability, even over the small distances. As can be seen in fig. 2, the character of this behavior is undoubtedly related to direction considered. In the research area spatial domain an elongated axis of anisotropy ellipse can be supposed in the direction of 45°, while in the orthogonal direction even the pure nugget effect can be observed (fig.2). As can be seen, the

variance stabilized at the distance of 500m and beyond this extent no significant autocorrelation is present.

### Temporal Variability Evaluation

Temporal variability evaluation has been based on a long term monitoring at 50 locations overlapping the part of research area as defined above (fig. 1). Till now 30 observations have been accomplished and the measurement still continues. In this stage we determined all descriptive statistics such as acceleration intensity, temporal autocorrelation coefficients, trend over respective period and residuals structure. For example, the most dynamic temporal behavior we observed was the transition from  $5\text{ kBq/m}^3$  to  $55\text{ kBq/m}^3$  during the period of 30 days (10 observations), what unambiguously moves this location to a higher radon risk category (with regard to the soil structure). Temporal variability spatial behavior and its relationship to the predicted radon field can be seen in figure. 3.

### Model Prediction

To predict unknown values in the research area spatial domain ( $D$ ) the *quasi-stationary ordinary kriging* has been used. The estimator is of the form

$$Z(x_0) = \sum_{i=1}^k \lambda_i Z(x_i) \quad \text{subject to} \quad \sum_{i=1}^k \lambda_i = 1$$

where  $Z(x_i)$  is  $k$  random variables inside  $d$  ( $ddD$ ),  $\lambda_i$  are individual weights. Detailed description of this method goes beyond the scope of this article and can be found in references. Since intensively skewed source data distribution the logarithmic transformation has been done. Consecutively will be processed only the logarithms of radon activity. Eventually, after the estimation the backward transformation is on hand. Resultant model overlaid by temporal variability isolines expressed in the relative standard deviation units (percentage of measured radon activity) can be seen in fig. 3.

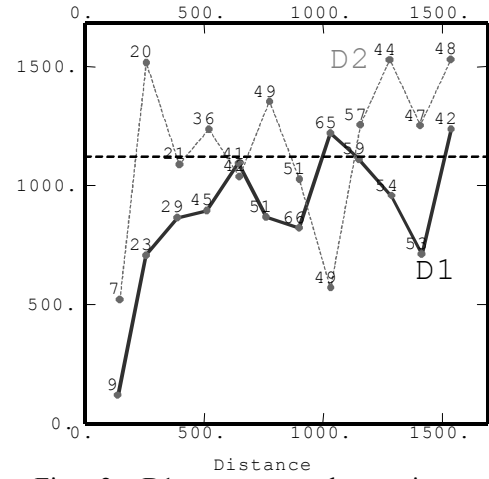


Fig. 2. D1 expresses the variogram constructed in the direction of maximal continuity, D2 expresses the variogram in the direction orthogonal to the first one. Figures express the numbers of pairs used for the respective variance calculation



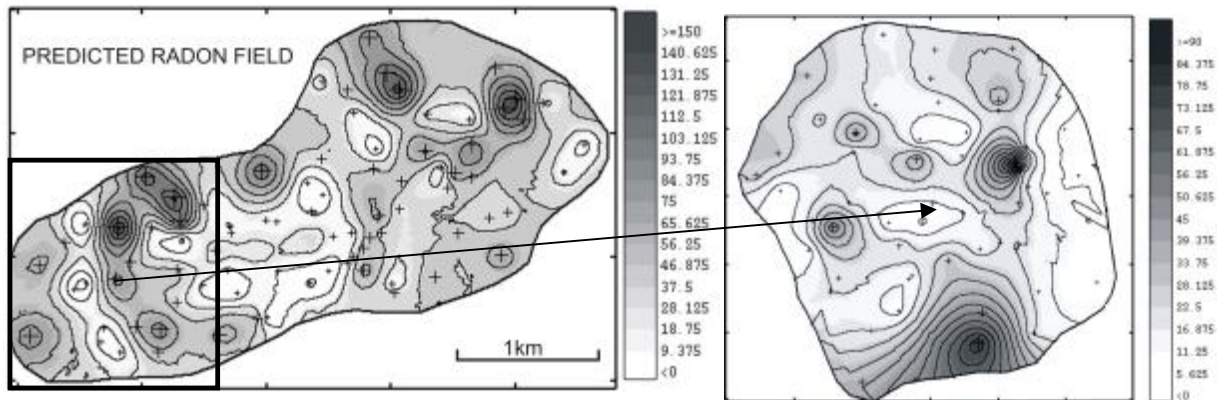


Fig. 3 Predicted radon field structure ( $\text{kBq/m}^3$ ) and respective model of temporal variability ( $s_x\%$ )

### Summary and discussion

Soil-gas radon potential assessment is an important component of the most of national radon programs. With regard to recent intensive advances in all fields of science and increasing demands on accuracy in data there is an urgent need to support current approaches by alternative methods. Important finding of this study is the benefit of variographic analysis based on random sampling for prediction of the overall radon potential, as a counterpart of a widely used regular sampling. In addition, as could be seen the temporal variability might be a crucial factor affecting consequent accuracy of radon risk assessment. We hope that introduced combination of geostatistics tools, results of long-term radon activity monitoring and, in general, dealing with uncertainties affecting radon potential/risk assessment can bring synergic effect providing more exhaustive data treatment.

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# INDOOR RADON EXPOSURE IN A SLOVAK RADON-PRONE AREA AND AN APPROACH TO THE ASSOCIATED HEALTH RISK ASSESSMENT

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## Introduction

Attention too the problem of exposure to radon indoors and the associated health has been growing all over the world in the both developed and developing countries. Until recent years radon was considered to be a radiation hazard only to the workers engaged in the mining and milling of uranium. This notion has dramatically changed in view of surveys carried out in many countries which showed high concentrations of radon in some dwellings that could entail significant health risks. The problems are more pronounced in elevated background radiation areas, buildings and houses built with materials containing significant levels of radium. The indoor radon problem is more pronounced in countries of the temperate zones where stringent energy conservation measures leads to tighter sealing of doors and windows during winter months. Assessment reported by the United Nations Scientific Committee on the Effects of Atomic Radiation in 2000 depicted that radon is major contributor to human exposure to natural sources, approximately 51.4 %. Radon indoors has given a new dimension to the perspective of the human exposure to radiation, both natural and man-made. It has also prompted many national authorities to adopt corrective measures, both regulatory and technical, in order to minimise exposure to radon indoors.

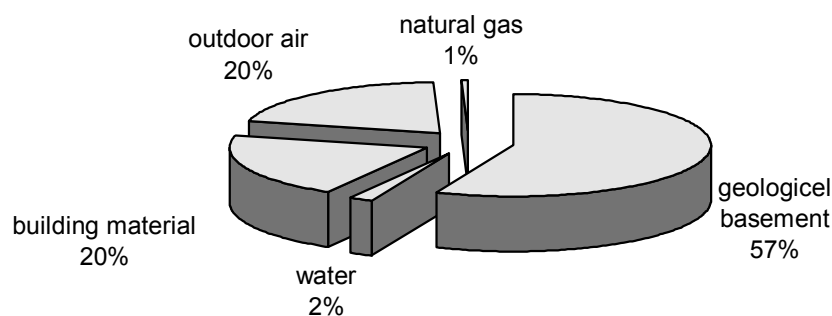


Fig. 1 Sources of radon input in to dwellings

Radon is classed as a human carcinogen on the basis of numerous studies of lung-cancer in radon exposed groups of underground miners. Five extensive epidemiological studies supported the mathematical modelling based on lung morphology and the inhalation characteristics of the progeny. These studies have shown a correlation between exposure to radon and excess lung cancer. With respect to smoking some studies on lung cancer in radon -

exposed miners suggest qualitatively a synergistic or multiplicative effect of smoking and radon exposure.

Slovakia belongs to the countries with higher risk of radon. The south-east part of Slovakia, called Spišsko-Gemerský region, represents the area where intensive underground activities took place, such as mining and milling of ores, rich of minerals and other resources. Due to the specific geological and tectonics structure, as well as to the presence of various anthropogenic pollutants in the investigated geochemical environment, elevated soil radon levels were found.

Local surveys for assessment of the average population exposure to indoor radon, carried out by our Institute, have shown strong variations in radon concentrations from one area to other, but in the observed area, elevated indoor radon concentrations were found in comparison to the Slovak national averages.



Fig.2 Ways of radon entrance in the building

Under the supervising of the Ministry of Environment the research work of several Slovak institutions was associated in the joint project, concentrated on the examination of the nature of contamination of the environment and the impact of available geochemical pollutants on the health status of the population in the identified area [1].

### **Material and methods**

In the Spišsko-Gemerský region it is situated about 47,555 residences (24,667 family houses and 22,888 multifamily dwelling units). The local indoor radon survey was until now realised in 349 dwelling units (it is 7.33% from the total number of dwellings). Meanwhile were indoor radon measurements performed only in about one third of municipalities.

From the total number of monitored homes (349) in 117 dwelling units the long-term radon concentrations exceeded the annual derived intervention level of  $500 \text{ Bq}\cdot\text{m}^{-3}$ , established in the new ordinance of Slovak Ministry of Health [2].

In spite of the fact that it was impossible to perform monitoring in all residences of the study subjects, we have made the attempt to analyse the radiation load of an individual living in the region, from indoor radon exposure, as well as to assess the annual average effective doses in the municipalities where more than 20 dwelling units were long-term monitored (Tab.1) .

Tab.1 : Assessment of the average annual effective doses of the population living in the Spišsko-Gemersky region from indoor radon exposure.

Municipality	Dwellings exceeding The action level of radon 500 Bq.m <sup>-3</sup>	Arithmetic mean of the Weighted radon concentration [Bq.m <sup>-3</sup> ]	E [mSv]
Poproč	14	460.0	7.9
Dobšiná	7	240.0	4.1
Rožňava + Čučma	26	117.5	2.0
Hnilec	23	735.0	12.6
Spišská Nová Ves	3	62.5	1.1

The weighted arithmetic mean of the radon concentration in the investigated region is 69 Bq.m<sup>-3</sup> and the mean effective dose due to radon exposure is 3 mSv per individual from population.

We suppose, that inadequate sample size as well as the interpretation of group-averaged exposure to radon as exposure to individuals, might be an important source of error. Therefore further objected extension of indoor radon measurements in the Spišsko-Gemersky region are expected for creation of reliable and equality of relevant data on indoor radon concentrations and for the analysis of the radiation load of population.

## Results

The mortality records for Slovakia support the significance of defining the risk of lung cancer from environmental exposure such as indoor radon and possible some other industrial fibrous dusts. Mortality due to the cancer is in second highest mortality rate in Slovakia [1]. In 1997 there were about 18,029 newly detected cancer cases, of which 9,943 were males (384 per 100,000 inhabitants) and 8,036 females (298.7 per 100,000 inhabitants). Lung malignancy had a dominant position in males, at a rate of 83.2 per 100,000 inhabitants (21.7% of total cancers). In females malignant breast tumours were in the first place, at a rate of 45.6 per 100,000 inhabitants (14.5%). Malignant lung tumours totally in both sexes were the most common cancer site, 2,436 cases (13.8%). Smoking is the main cause of lung cancer, the health statistics revues that there are 27% males and 13% females who smoke regularly in Slovakia. The next leading causes of lung cancer are believed to be indoor radon, with an attributable risk of 5-20 % of all lung cancer.

Table 2 Assessment of health risk from indoor radon exposure.

Region	Radon concentration [Bq.m <sup>-3</sup> ]	Risk assessment*
Slovensko	120	11.48
SG región	172.5	16.51
Spišská N.Ves	160	15.31
Košice-okolie	185	17.70
Geľnica	172.5	16.51
Rožňava	250	23.92

\* supposed additional lung cancer deaths for 100 000 inhabitants exposed to radon

For estimating the risk of indoor radon the BEIR VI model have been used. The estimate considered separately exceded long cancer risk for never smoker and ever smokers. The results of the risk analysed is given in the table 2 and fig. 4. The number of measured dwellings which exceed the action level 500 Bq.m<sup>-3</sup> in various regions of SGR are given in fig.5.

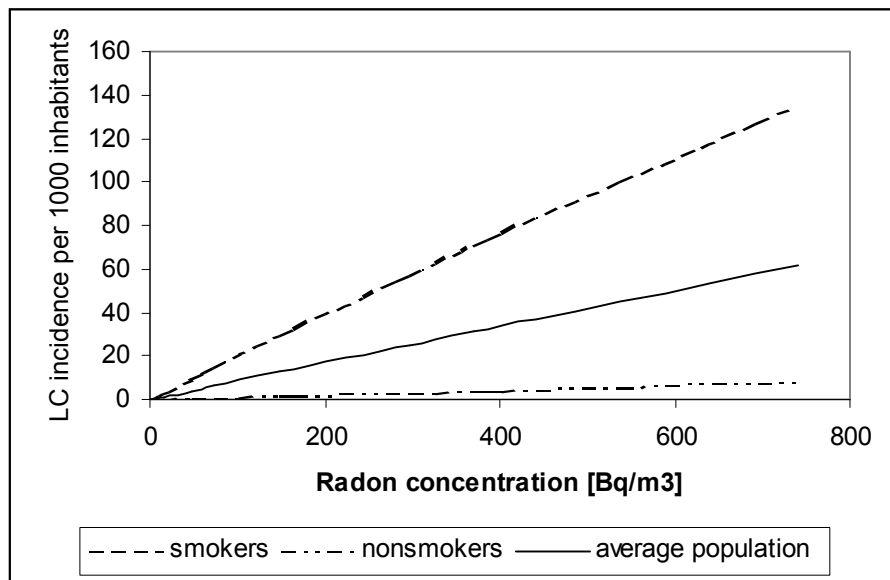


Fig. 3 Trends of lung cancer incidence in Slovakia, in the years 1968 - 1998ings where radon screening was realized in various parts of SGR.

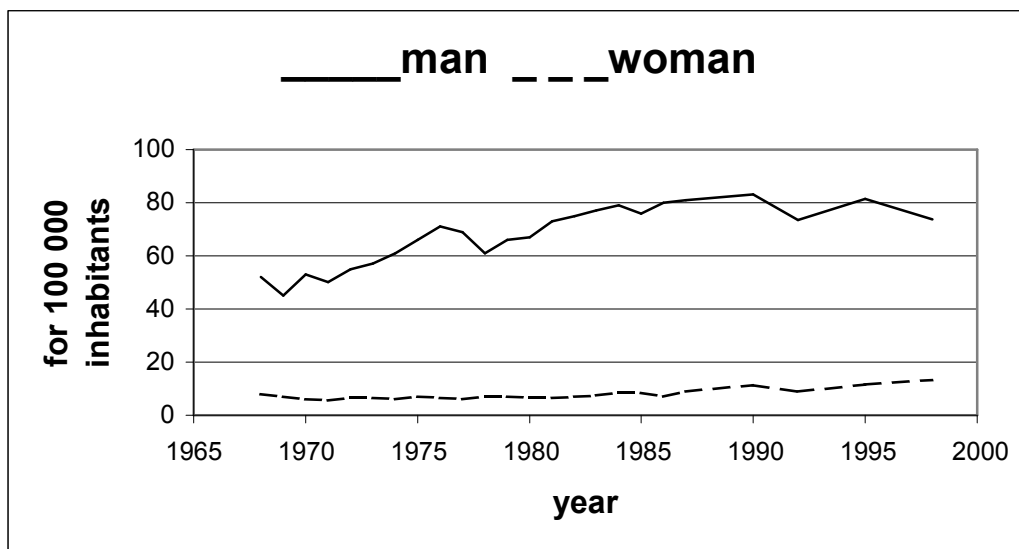


Fig.4 Lung cancer incidence among smokers, nonsmokers and average population exposed to radon

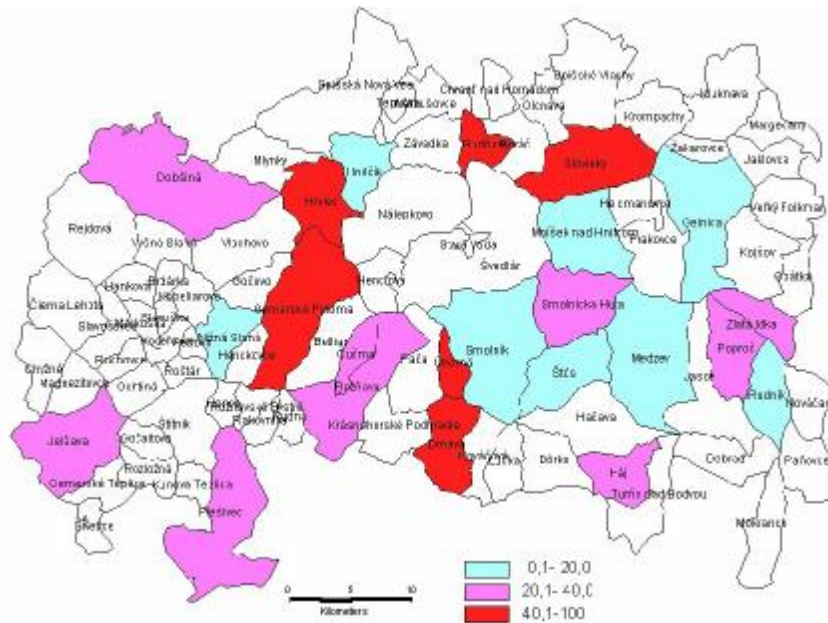


Fig.5 The number of dwellings in regions of SGR exceeding the action level

## Conclusion

The knowledge of the impact of radon exposure on the health status of the population, in spite of great uncertainties, contributes to understanding the associated risk and helps to implement remedial program for radon exposure reducing. Average radon exposure does not determine the average risk, but extrapolation of the data observed in cohorts of underground miners to the lung cancer risk at low radon progeny exposure is most successful. Further objected extension of Rn measurements are needed for creation of reliable set of relevant data on radiation load of population in SGR.

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# **$^{210}\text{Po}$ - AND $^{210}\text{Pb}$ - DETERMINATION IN HUNGARIAN TOBACCO LEAVES AND DOSE ASSESSMENT**

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## **Introduction**

The extremely harmful carcinogen effect of smoking is well known. Besides other chemical agents, the radionuclide content of the tobacco is not negligible either<sup>1</sup>. Among the radioactive isotopes in tobacco,  $^{210}\text{Po}$  is one of the most remarkable, because at the lighting temperature of a cigarette  $^{210}\text{Po}$  compounds are volatile enough to go with the smoke to the respiratory tract.

The other important isotope is  $^{210}\text{Pb}$  being the mother element of  $^{210}\text{Po}$ , thus continuously produces its daughter<sup>2</sup>.

The fact that these radionuclides can enrich in the tobacco leaves has been emphasised by several authors assuming that the big surface of the leaves (caused by the special leaf-hairs)<sup>3</sup> and the long growing time of the plant (long absorption period for the isotopes) could be the explanation<sup>4,5</sup>.

Considering the incidence and the harmful effect of smoking, it is a serious endemic. Consequently, it is important to determine the concentrations of  $^{210}\text{Po}$  and  $^{210}\text{Pb}$  in the different kinds of cigarettes. According to surveys, 36-39% of the adult population and 18-22 % of the youngsters under 18 smoke<sup>6</sup>.

Adults smoking regularly have 16 cigarettes daily on average but a high number of people smokes more than 40 per day.

Because the tobacco of the cigarettes sold in Hungary are domestically grown; the aim of our investigations was to determine the concentrations of  $^{210}\text{Po}$  and  $^{210}\text{Pb}$  in the tobacco plants coming from the different areas of Hungary.

Additionally, soil samples and phosphor fertilisers as the potential sources of these radionuclides were also investigated.

Special attention was given to the tobacco grown at a tailing pond of a former uranium mine under remediation and the plants grown nearby. One of the biggest tobacco plants is situated there.

## **Materials and methods**

### **Sampling**

The  $^{210}\text{Pb}$ - and  $^{210}\text{Po}$  activity concentration was investigated in tobacco leaves, root, stem, soil, and the most popular phosphate fertilisers. The tobacco samples were collected from tobacco plants situated in different regions of Hungary including a place close to a remediated uranium mine. The soil samples came from the same places and the phosphate fertilisers were collected from 5 different companies. The tobacco samples were collected in 2002; we have none from 2003 yet.

### **Leaching of $^{210}\text{Po}$ from phosphate fertilisers:**

The solubility of phosphate fertilisers in citrate and water were measured. The fertiliser samples were dried at  $110^\circ\text{C}$  for 24 hours then 5g were put into  $250\text{ cm}^3$  2,6 mol/L ammonium citrate solution and de-ionized water, respectively. After 1, 6 and 24 hours of continuous stirring, the sample was filtered. The  $^{210}\text{Po}$  concentration was determined from the solution.

### **Determination of $^{210}\text{Po}$ concentration:**

A known amount of  $^{208}\text{Po}$  tracer was given to 2g of dried sample, then 3x20 ml cc  $\text{HNO}_3$  was added. Afterwards it was almost boiled down and with 3x30 ml cc  $\text{HCl}$  the boiling process was continued in order to give away the nitrates. (In the case of soil samples first 3x30 ml  $\text{HF}$  boiling down was executed with continuous stirring.)

The source preparation was carried out by spontaneous deposition on acid resistant plate with high Ni content<sup>2,7,8</sup>.

### **Determination of $^{210}\text{Pb}$ concentration:**

Following the determination of  $^{210}\text{Po}$  concentration the solution was stored for 6 months then the  $^{210}\text{Po}$  was measured again as described above. Using this result the  $^{210}\text{Pb}$  concentration was calculated by the ModelMaker software.

## **Equipment**



The measurement of  $^{210}\text{Po}$  was carried out with alpha-spectrometry array formed by three chambers (Tennelec TC256, Eurisys Mesures 7184 and Canberra Model 7401) working in parallel. Each chamber had a Passivated Implanted Planar Silicon (PIPS) detector with resolution 19 keV (Eurisys Mesures), using an Oxford Tennelec PCA Multiport Analyzer and Oxford Tennelec DMR108A Multiplexer.

## Results and discussion

*Table 1.*

*The  $^{210}\text{Po}$  concentration of cigarette samples and the consequent effective doses*

Sample	$^{210}\text{Po}$	Eff.	Sample	$^{210}\text{Po}$	Eff.	Sample	$^{210}\text{Po}$	Eff.
	act./spear [mBq]	doses [mSv]		act./spear [mBq]	doses [mSv]		act./spear [mBq]	dose [mSv]
A11	30.8	0.16	A24	43.2	0.22	B22	15.4	0.08
A12	30.5	0.15	A25	39.8	0.20	B23	12.1	0.06
A13	33.3	0.17	A31	51.3	0.26	B24	18.8	0.09
A14	33.5	0.17	A32	50.0	0.25	C11	26.4	0.13
A15	31.5	0.16	A33	53.7	0.27	C12	38.7	0.19
A16	30.6	0.15	B11	31.3	0.16	C13	32.5	0.16
A17	32.1	0.16	B12	33.8	0.17	C14	34.0	0.17
A21	40.0	0.20	B13	32.8	0.16	C15	29.9	0.15
A22	40.0	0.20	B14	33.5	0.17	C16	28.9	0.14
A23	42.3	0.21	B21	13.2	0.07	-	-	-

In Table 1. A, B, and C mean the tobacco company, the first number after the letters means the brand of the cigarette and the second one the type (ex. lights, ultra lights). It can be seen that the  $^{210}\text{Po}$  concentration is the same in the case of a given brand and is not affected by the tar and nicotine content <sup>9</sup>.

Here can be seen the assessed dose contribution as well. At the calculation  $4.3 \cdot 10^{-06}$  Sv/Bq dose conversion factor<sup>10</sup> and 20% of the whole  $^{210}\text{Po}$  depositing in the lung<sup>11</sup> with 16 cigarettes per day was considered. The assessed annual effective dose due to regular smoking is between 0.07 and 0.26 mSv, which can present a relatively considerable exposure. In the case of smoking two boxes per day it can reach 0.68 mSv/year.

Figure 1. presents  $^{210}\text{Po}$  (7-195 mBq/g) and  $^{210}\text{Pb}$  (4-32 mBq/g) activity concentrations of different parts of tobaccos grown in Hungary.

It can be seen that the  $^{210}\text{Po}$  concentrations are between 7-195 mBq/g and 4-32 mBq/g for  $^{210}\text{Pb}$ . These results are similar to the ones measured in other countries. The exception was the tobacco grown at the remediated tailing pond where the  $^{210}\text{Po}$  concentration measured in

the leaves was 1500 mBq/g and 784 mBq/g in the soil. Figure 1. also shows that the leaves contain the highest amount of  $^{210}\text{Po}$  and  $^{210}\text{Pb}$ , presumably due to the aerosol deposition on the big surface. The radionuclide concentration of the samples came from areas where the soil radionuclide concentration is also high. (Sample 3 came from a plant close to an uranium mine).

The measurements show that the  $^{210}\text{Po}$  and  $^{210}\text{Pb}$  concentrations of the tobaccos and cigarettes do not differ considerably.

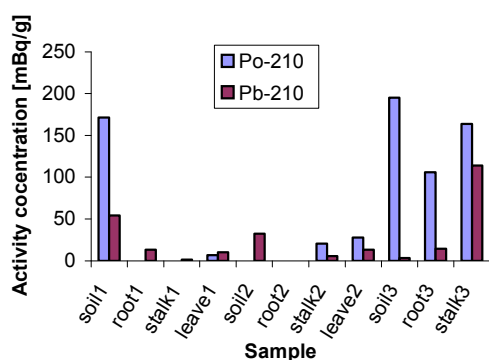


Fig. 1. The  $^{210}\text{Po}$  and  $^{210}\text{Pb}$  act. conc. of different parts of tobaccos

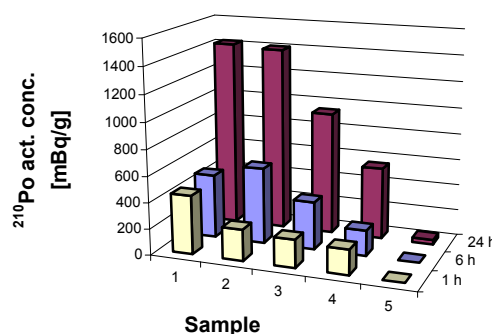


Fig.2. Leaching of  $^{210}\text{Po}$  from phosphate fertilizers in ammonium citrate medium

According to some authors, the high concentrations of these two radionuclides are due to phosphate fertilisers. Therefore, the  $^{210}\text{Po}$  concentrations of the phosphate fertilisers available in Hungary and the chemical form of these radionuclides, namely, how can the plants built them in, were also investigated.

Figure 2 shows the leaching of polonium from phosphate fertilisers in ammonium citrate medium. The 5 samples come from different factories. It can be seen that the leaching of  $^{210}\text{Po}$  depends on the time considerably.

There was no detectable result leaching in de-ionized water.

## Conclusion

The  $^{210}\text{Po}$  and  $^{210}\text{Pb}$  concentrations of the investigated tobaccos and cigarettes do not differ considerably, so during the cigarette making process the concentration of these radionuclides does not change to a great extent.

The  $^{210}\text{Po}$  concentrations of the phosphate fertilizers available in Hungary were not found extremely high in either case. The ( $^{210}\text{Po}$  72-1498 mBq/g,  $^{210}\text{Pb}$  313-985) concentrations of the tobacco grown at the experiment plant situated at a former uranium mine (waste hip) were high, as expected. This territory is not available for agricultural purposes. Our results also indicate that it is reasonable.

The samples from the totally covered (remediated) tailing pond are under investigation.

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# BEHAVIOUR OF $^{222}\text{Rn}$ AND ITS DAUGHTER PRODUCTS IN OPEN ATMOSPHERE

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## **Introduction**

The radon ( $^{222}\text{Rn}$ ) is formed in the ground and released into the atmosphere. After the radioactive decay the radon progenies are rapidly fixed on the aerosol particles of the air. The transport of the radon and its decay products in the troposphere is mainly determined by the turbulent diffusion. Besides the diffusion the concentration profiles are influenced by the decay of the radionuclides and, in case of the decay products, by the wet and dry removal of aerosols from the atmosphere. In the last years the extensive research of the behaviour of radon and radon progenies in the outdoor air has been made. It has been reported by various authors that atmospheric concentrations of  $^{222}\text{Rn}$  and its short-lived daughters show the diurnal variations with maxima early morning and minima in the afternoon [1-3].

In the last period an important motivation for the study of the radon in the outdoor atmosphere is using of the  $^{222}\text{Rn}$  and its decay products in the atmospheric studies, especially for the determination of the atmospheric stability [4, 5]. The new knowledge about behaviour of the radon and radon daughters in the atmosphere are needed also for more precise determination of public radiation exposures from radon.

In this contribution the influence of the meteorological conditions on activity concentrations of the outdoor  $^{222}\text{Rn}$  and its daughter products is discussed in detail. In addition, the correlation between concentrations of measured radionuclides is studied and empirical relations for concentrations of radon daughter products are presented.

## Methods

The measurements of radon and its short-lived daughters were carried out on the open grass area in the campus of the Faculty of Mathematics, Physics and Informatics in Bratislava. The air for the analysis was sucked at the height of 1,5 m above the ground surface.

The radon activity was continuously monitored using the large volume scintillation chamber [6]. The radon monitor allows the measurements of the radon activity concentrations in the surface layer of the atmosphere with uncertainties less than 30 % in the counting interval of 2 hours.

The activity concentrations of  $^{218}\text{Po}$ ,  $^{214}\text{Pb}$  and  $^{214}\text{Bi}$  were measured continuously by Si(Au) detector. The decay products of  $^{222}\text{Rn}$  were collected from the sucked air on the filter surface and then the filter's alpha activity was spectroscopically analysed in three time intervals [7]. The uncertainties of the radon daughter measurements were as a rule less than 20 % at the flow rate of the air through the filter equal to  $50 \text{ l}\cdot\text{min}^{-1}$ . The data were collected with the frequency of 2 hours.

## Results and discussion

The behavior of the radon and its short-lived decay products in the open atmosphere was systematically investigated during the period of time March – November 1998.

The daily variations of radon and its decay products in the outdoor air are illustrated in Fig.1. We chose the time interval from 14 October to 10 November, because in this period there were well observed the changes of the radon and its decay product concentrations under the different atmospheric conditions. At first sight there can be seen the good agreement between courses of concentrations of the  $^{222}\text{Rn}$  and its decay products. Moreover, in some details both the courses are identical, for example the increasing of the activity concentrations for short time (see 26<sup>th</sup> October and 9<sup>th</sup> November). The influence of the wind velocity on the activity concentration was observed in all courses. The increase of wind velocity is attended by the decrease of the activity concentration. The regularly changing wind velocity during a day is attended by the daily variations of the activity concentrations of the  $^{222}\text{Rn}$  and its decay products. This effect is clearly seen during days from 15<sup>th</sup> October to 20<sup>th</sup> October. Further we can see days when neither significant variations of the wind velocity nor expressive variations of the activity concentrations were observed (20<sup>th</sup> October – 24<sup>th</sup> October). However the intense increasing of the wind velocity led to the decrease of the activity concentrations from 29<sup>th</sup> to 30<sup>th</sup> October. After 1<sup>st</sup> November the wind was characterised by the moderate decrease of the average velocities and by their small daily variations. The activity concentrations

slowly increased and the daily variations were not distinct. Naturally the other processes (like cloudiness) also play the role in the courses of the activity concentrations.

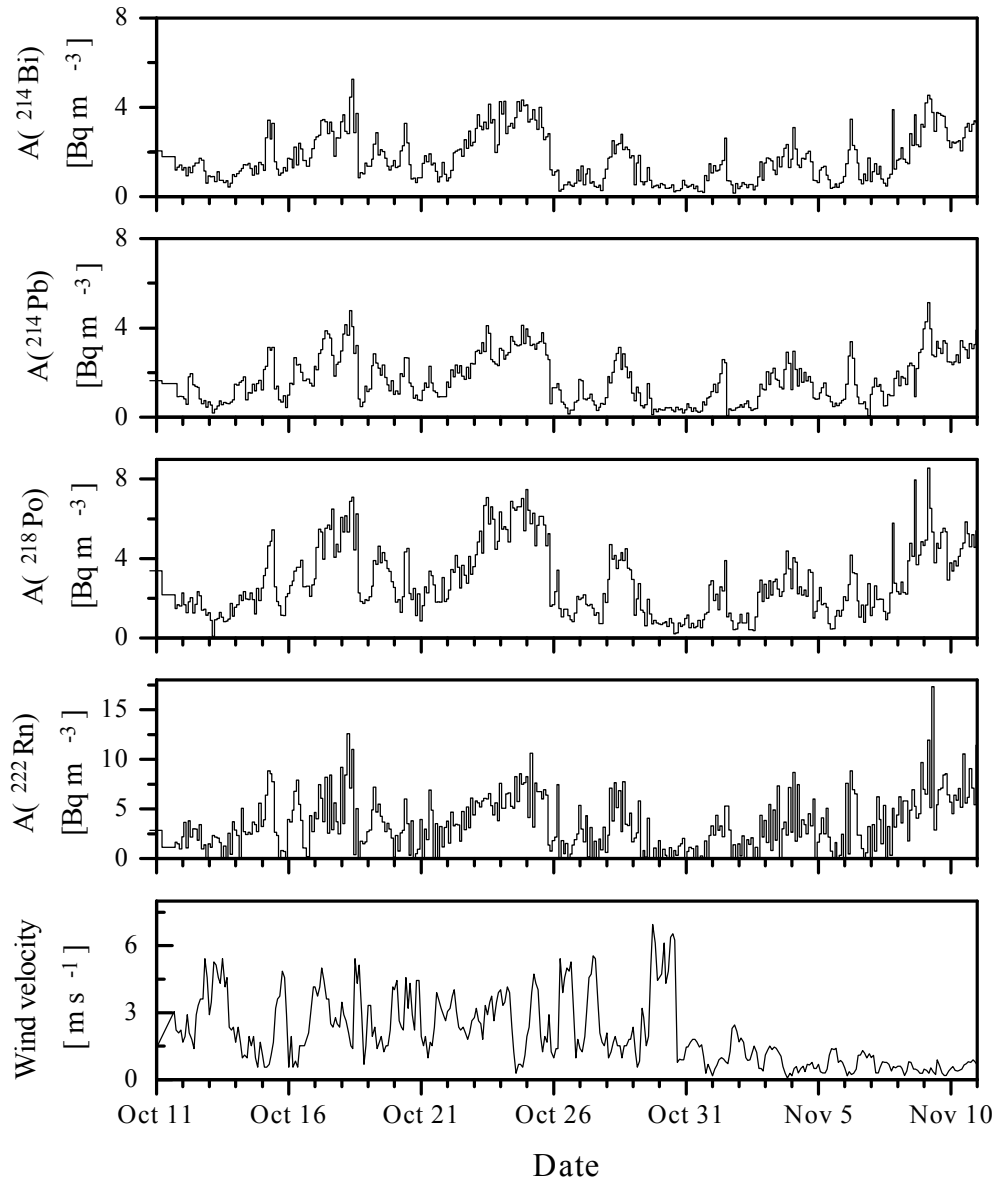


Fig. 1. The activity concentration of radon and its daughter products in the atmosphere.

On the basis of the measured data there have been studied the dependences of the concentrations of short-lived decay products on the radon concentration particularly as well as the behavior of the equilibrium factor  $F$  in the atmosphere. For the autumn period the following relations for the activity concentrations of daughter decay products of the radon in the surface layer of the atmosphere were determined (Fig. 2):

$$A_{218Po} = 1,06 (A_{222Rn})^{0,79}, \quad A_{214Pb} = 0,57 (A_{222Rn})^{0,91}, \quad A_{214Bi} = 0,62 (A_{222Rn})^{0,86},$$

where  $A_{222Rn}$  is radon activity concentration.

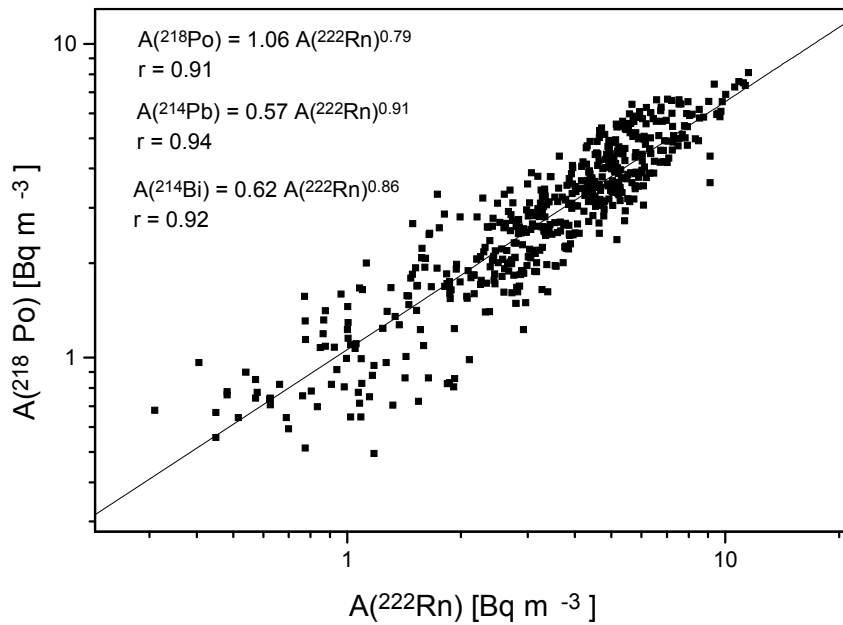


Fig. 2. Correlation between  $^{218}\text{Po}$  and  $^{222}\text{Rn}$  activity concentration in the atmosphere.

The mentioned dependences are not linear. The ratios of the activity concentrations of the short-lived decay products to the radon activity concentration decrease with the increasing radon activity concentration. The similar dependence was determined by G. Keller [8] and we confirmed it using five times bigger data set.

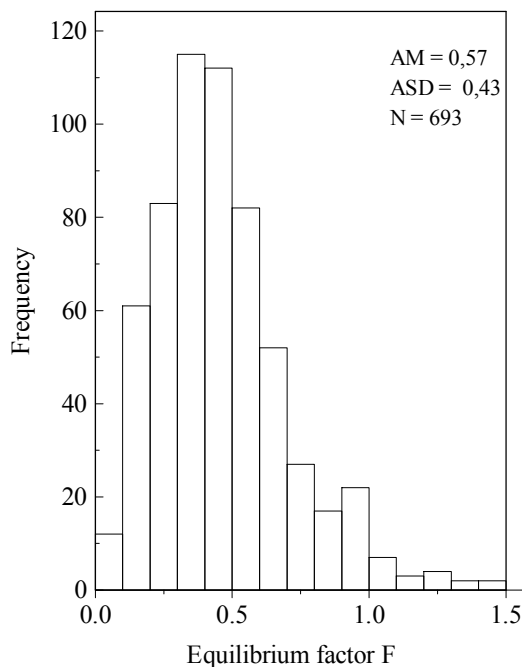


Fig. 3. Frequency distribution of equilibrium factor  $F$  in the atmosphere.

The time course of the equilibrium factor  $F$  in the surface layer of the atmosphere ranges from 0,3 to 1,5 with the average value 0,57 (Fig.3). As the measurement results showed the equilibrium factor strongly depends on the meteorological conditions and mostly on the wind velocity. At the wind velocity about  $6 \text{ m}\cdot\text{s}^{-1}$  there were measured the biggest values of the equilibrium factor and its value was sometimes bigger than 1. The explanation of this “abnormal” ratio is beyond the possibility of the simple atmospheric diffuse model [9]. Such situation can arise for example when the decay products of radon are transported horizontally on the sampling place from the other localities.

## Conclusion

The courses of the activity concentrations of  $^{222}\text{Rn}$  and its short-lived decay products in the outdoor atmosphere are very similar even in some details.

The average equilibrium factor  $F$  determined on the basis of our measurements in the outdoor atmosphere is approximately about 20 % lower than the value  $F$  recommended by UNSCEAR 1996 [10] for the open air ( $F = 0,7$ ).

The obtained nonlinear relations between radon and its decay products reduce the applications of decay products in the atmospheric studies on one hand but on the other hand they can be useful at the calculation of the equilibrium equivalent radon concentration with the aim to determine the effective radon dose on the basis of the measurements of its activity concentrations only.

## Acknowledgements

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# STUDIES ON THORON PROGENY IMPLANTATION IN DIFFERENT MATERIALS

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## **Introduction**

Many retrospective studies regarding the time exposure to radon ( $^{222}\text{Rn}$ ) are based on the accumulation of  $^{210}\text{Pb}$  ( $T_{1/2}=22.3\text{y}$ ) on the surface of glass objects or other materials and measuring the specific alpha activity of  $^{210}\text{Po}$  ( $T_{1/2}=138$  days,  $E_{\alpha}=5.3$  MeV) resulted by implanting of radon daughters, due to the alpha recoil [1-3]. For thoron ( $^{220}\text{Rn}$ ) this method is not applicable because there is not a long lifetime isotope to allow a significant accumulation of any descendant. The longest lifetime of the thoron descendants is  $^{212}\text{Pb}$  with  $T_{1/2}= 10.6$  hours. But a study involving thoron daughters implanted at the surface of glasses or other materials can be useful for the simulation of radon progeny implantation. This simulation is possible due to the equilibrium of implantation, which is obtained in a short time (~two days) compared to about 100 years in the case of  $^{222}\text{Rn}$ . An alpha spectrometry method was recently applied to measure the thoron contribution in the case of integrating track-etch detectors used for thoron monitoring or in the studies regarding radon and thoron progeny deposition. The purpose of this work is to obtain the alpha spectra for thoron daughters implanted at the surface of different materials and to show the possibility of this method to simulate the radon progeny implantation and thus to find some parameters used in the Jacobi model [4].

## **Experimental method**

The obtaining of the alpha spectra and the alpha counting were made using a PIPS alpha-detector of Canberra type ( $900\text{ mm}^2$ ) with 25 keV energy resolution operated under

vacuum conditions and connected to a pulse height analyzer type. For alpha energy calibration we used the internal standard pulses generated by the alpha spectrometer. The detection efficiency was calculated taking into consideration the average solid angle from which the detecting surface ( $900 \text{ mm}^2$ ) is seen from sample surface ( $100 \text{ mm}^2$ ) [5]. The sample is placed on the detector axis. The efficiency calculated in this way was 35.5%.

Two hundred grams of  $\text{Th}(\text{NO}_3)_2 \cdot 5\text{H}_2\text{O}$  powder were put into a metallic box of 8.5 cm diameter and 5.0 cm height, Fig.1. This substance was purchased over 25 years ago, therefore the equilibrium between  $^{232}\text{Th}$  and  $^{234}\text{Ra}$  is almost reached (>95%). A thin plastic sieve was placed below the thorium nitrate. The thickness of  $\text{Th}(\text{NO}_3)_2 \cdot 5\text{H}_2\text{O}$  layer was 2.9 cm and that of the layer of air above was 2.1 cm. Taking into account the very short live time of thoron (1 minute), its equilibrium concentration is quickly reached in  $V_2$  ( $164 \text{ cm}^3$ ) after closing the box. We considered the time of maximum 30 minutes, which also allows the diffusion of the thoron generated from  $V_2$  into  $V_1$  ( $134 \text{ cm}^3$ ). The amount of thoron in volume  $V_1$  was determined by isolating the thorium nitrate in a 0.5 L plastic bottle hermetically closed, the thickness of thorium nitrate being the same as in the box from Fig. 1.

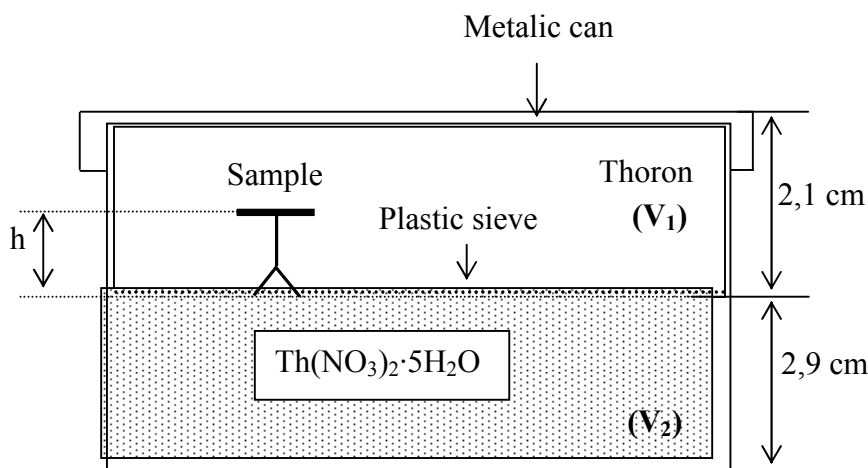


Fig 1. The container for the irradiation of samples with thoron

## Results and discussion

Fig.2 shows the alpha spectrum obtained for the glass sample. Only the peaks at the energy of 6.05 MeV and 8.78 MeV are the real peaks. The first peak is obtained from alpha disintegration of  $^{212}\text{Bi}$  (36%) and the second from  $^{212}\text{Po}$  disintegration generated by the beta disintegration of  $^{212}\text{Bi}$  (64%). The 6.5 MeV, 7.5 MeV and 8.5 MeV peaks from this spectrum are peaks generated by the alpha spectrometer standard impulse generator, in order to be used

at energy calibration. If the energy level is used for the energies  $E_1$  5 MeV and  $E_2$  7 MeV, the areas ( $N_1$  and  $N_2$ ) of these two peaks were obtained, and the weighting factor can be found.

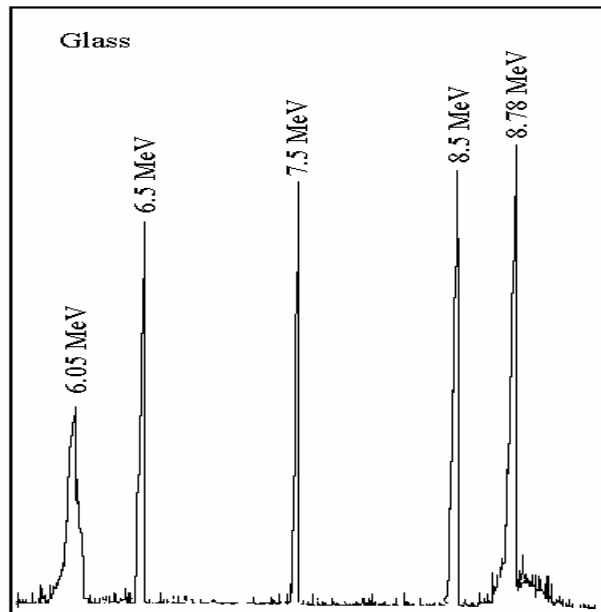


Fig 2. The alpha spectrum obtained for the glass sample

As this spectrum shows, the peaks are relatively narrow, especially the 8.78 MeV peak. The width of the peaks strongly depends of the material also of the surface roughness. Table 1 presents the implantation data obtained from 7 types of materials. As it can see from this table, the weighting factor (64%) for beta decay of  $^{212}\text{Bi}$  was found. The average value for our results is  $63.25 \pm 0.95$  % as can observe from the third column from this table. The width  $\Delta E$  of the lines depend on the material used and the surface processing. The narrowest lines were recorded when we are working with glass and plexiglas having a very glossy surface.

Table 1. Some characteristics of thoron progeny implanted in different materials.

Nr	Sample material	g(8.78MeV) %	DE (keV)	$N_1$ (imp/100s)	$f_i$ (Bq/m <sup>2</sup> :Bq/m <sup>3</sup> )	Obs.
1	Glass	63.7	55	1080	0.062	
2	Polyethylene	61.8	64	672	0.039	
3	Paper	62.6	89	1010	0.058	glossy
4	Plexiglas	65.1	46	355	0.020	
5	Cu	63.3	137	620	0.036	polished
6	Cu+ HNO <sub>3</sub>	63.2	221	862	0.049	
7	Al	63.5	146	175	0.010	foil

Taking for glass and Plexiglas an average width of 50 keV at the 8.78 MeV line and considering that the alpha protective cover of the detector introduces a supplementary width of the line of 10 keV (from 25 keV to 35 keV), we can calculate that the excess width of 15 keV is the result of a depth implantation of about 100 nm for  $^{212}\text{Pb}$ . Compared with other obtained data this value is almost 50% greater [6]. The excess width for the Cu sample treated with  $\text{HNO}_3$  (221keV) suggests that the implantation of thoron and radon progeny could be used in the study of surface roughness.

The last column in Table 1 shows the equilibrium implantation factor. This factor is defined as the implanted activity on surface unit ( $\text{Bq/m}^2$ ) divided by thoron concentration in the chamber ( $\text{Bq/m}^3$ ). Big values of the implantation factor are remarked in the case of the glass and the paper which is over six times higher compared with the Al sample. The value found for glass in the case of equilibrium for Thoron ( $^{212}\text{Pb}$ ), ( $65 \text{ Bq/m}^2/(\text{kBq/m}^3)$ ) is inough different of those found experimentally for Radon ( $^{210}\text{Pb}$ ), (about  $25 \text{ Bq/m}^2/(\text{kBq/m}^3)$ ) [7]. This difference can be explained by the important losses due to the corrosion process and diffusion during time of  $^{210}\text{Pb}$  to the glass surface as was remarked by Cauwels and Poffijn [2].

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# MEASUREMENT OF AVERAGE RADON GAS CONCENTRATION AT WORKPLACES

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## **Introduction**

Reference level for radon concentration in workplaces was introduced in several countries, including Hungary from 01. January 2003, following the EU suggestion [1]. The determination of average radon concentration could present a problem considering, for example, that the monthly values measured in a hospital cave showed 24 times difference depending of the chosen month [2]. In cases like this only the year-long measurements give reliable results [3]. There is a difference between the averages measured during the working hours and during the total time (including nights and weekends), mostly in the cases of rooms with frequent air change like the schools and kindergartens and the ventilated workplaces.

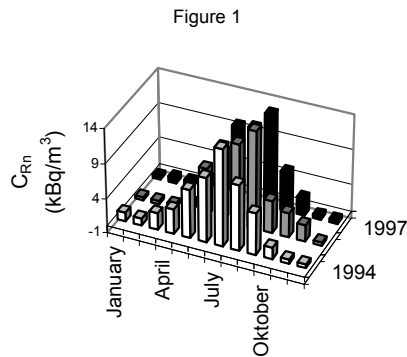
## **Materials and methods**

*a.) The effect of measuring duration.* To find out the ideal measuring length of time the radon concentration was measured continuously (hourly averages) in the hospital cave of Tapolca for three years long. The device was a Dataqua apparatus with semiconductor detector, which keeps the hourly averages in the memory.

*b.) The effect of considering the actual working time.* The ratio of the average radon concentrations measured during the actual working time and during the a whole time period were investigated in cases of schools, kindergartens, a uranium tailing pond and the hospital cave above mentioned. The measuring devices were: a PYLON-AB5 with CPRD scintillation detector.

## Experimental results

a.) *The effect of measuring duration.* The monthly average radon concentrations during



three years in the hospital cave can be seen on the Fig. 1. It can be seen that in the hospital cave the variation of radon concentration with the seasons is considerable. The average radon concentrations as a function of measuring duration (1, 3, 6 or 12 months) and measuring date can be seen in the Table 1.

Table 1

Measuring duration (month)	Average radon concentration (kBq/m <sup>3</sup> )			
	Minimum		Maximum	
1	0.51	0.03	12.4	2.6
3	0.67	0.23	11.54	0.63
6	1.11	0.38	8.23	0.35
12	4.67 0.12			

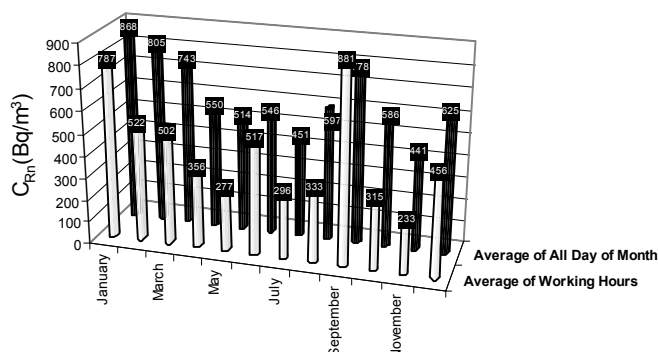
The values mean the differences of the certain year averages from the 3 years average.

It can be seen that at the one-month-long measurement in December the average was 0.51 kBq/m<sup>3</sup> and in August 12.4 kBq/m<sup>3</sup>, which means 24 times difference. Even when one measures in, 7 times differences could be resulted. If the six-months-long period measurements executed from the first and second part of the year (01. January – 30. June, and 01. July – 30. December) the averages are 3.64 0.37 kBq/m<sup>3</sup> and 5.7 0.40 kBq/m<sup>3</sup> consequently. These differ from the yearly average (4.67 0.12 kBq/m<sup>3</sup>) with approximately 20 %, which is acceptable. Consequently, in workplaces where the expectable changes of radon concentration considerable with the seasons should be measure for 12 months long. If it is not possible, the chosen six months period should contain summer and winter months as well.

It can be seen that at the one-month-long measurement in December the average was

b.) *The effect of considering the actual working time.* Analysing the results of

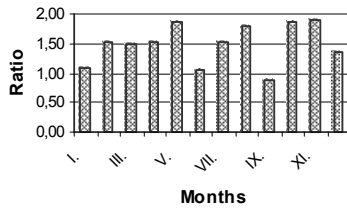
Figur 2



measurements in the hospital cave, it was found that the yearly average radon concentration was 17 % lower than the all day average in the morning hours, when the treatments happened, but 16 % higher for ones who spend 8 hours in the treatment

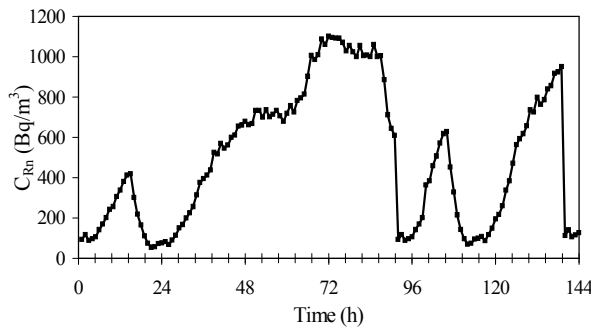
rooms of cave. This means that the average is higher during the working hours, but the

Figur 3



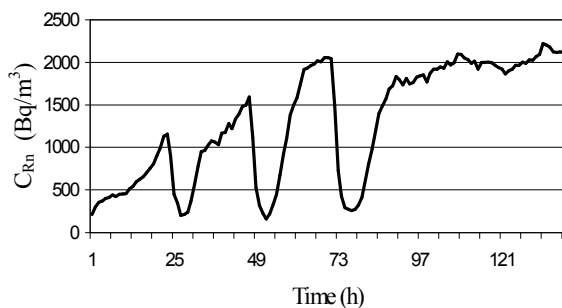
difference is not considerable. The monthly average radon concentrations (in 2001) during the working hours and the total time in a case of a uranium tailing pond when recultivation works happened, can be seen in the Fig. 2. The yearly average for the whole time was  $625 \text{ Bq/m}^3$ , which is high considering that this is an open-air place. Analysing the certain months the differences are well-marked. The ratio of the averages of the working time and the total time (Figure 3.) are changing with 40 %. The works connected with intensive radon exhalation and the meteorological circumstances causes this. This was the reason that in September the average for working hours was higher than the average for whole month. At the uranium tailing pond was find that

Figur 4



the personal dosimeters showed two times higher dose than the detectors placed to the workplace. The reason was that the personal detectors were stored in the changeroom where the radon concentration was higher than the on the workplace. This also point out that consider the exact working hours could present a problem at the integrated measurements. Analysing the results of measurements in schools other differences were found. The Fig.4. shows the results of measurements happened in a school. It can be seen that the radon concentrations increase during the nights and weekends, so the average is  $497 \text{ Bq/m}^3$ . During the time when the children are inside, the averages are  $125 \text{ Bq/m}^3$ .

Figur 5



Consequently, the average radon concentration during the effective time is only half of the most rigorous regulation ( $150\text{-}200 \text{ Bq/m}^3$ ) while the averages concerning the whole time exceed that. Similar results were found in the other cases as well. So, it would be unnecessary to mitigate in these cases because the levels for the effective time, usually, considerable less than the levels for the whole time. Basing on our experiences it can be stated that it would be

advisable to execute short time (a week) continuous measurements parallel with the long time (some months) ones to find out the variation of the radon concentration as well. At workplaces where the ventilation is effective or there is intensive air change due to the frequent opening the doors and/or windows, the averages for the working hours considerable differ from the averages of whole time. The Fig.5. shows the change of radon concentration in a mine tunnel. The ventilation works only during the working hours, so the levels increase at nights and in the weekends while during the working hours considerable lower. For a month track detectors were placed to the different locations of the mine and some miners wore the same detectors during work, which were keep in a place with low radon concentration ( $< 12 \text{ Bq/m}^3$ ) after the working hours. The average radon concentration in the mine was  $3690 \text{ Bq/m}^3$ , in working time was  $815 \text{ Bq/m}^3$  which means 4.5 times difference. The artificial ventilation effects strongly the radon concentration during working hours so, mitigation action should be done knowing the actual levels during the effective work time.

### **Conclusion**

The facts above detailed show that much more disturbing effects emerge at the determination of average radon concentration in workplaces than in cases of homes. It can be stated that the one month long measurements means very high variation (as it is obvious in the cases of the hospital cave and the uranium tailing pond). Consequently, in workplaces where the expectable changes of radon concentration considerable with the seasons should be measure for 12 months long. If it is not possible, the chosen six months period should contain summer and winter months as well. The average radon concentration during working hours can be differ considerable from the average of the whole time in the cases of frequent opening the doors and windows or using artificial ventilation.

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# THE ESTIMATION OF EFFECTIVE DOSES USING MEASUREMENT OF SEVERAL RELEVANT PHYSICAL PARAMETERS FROM RADON EXPOSURES

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## **Introduction**

The naturally radioactive gas radon ( $^{222}\text{Rn}$ ) is present in the air outdoors and in all buildings, including workplaces. Inhalation of the short-lived radon decay products ( $^{218}\text{Po}$ ,  $^{214}\text{Pb}$ ,  $^{214}\text{Bi}/^{214}\text{Po}$ ) yields the greatest amount of the natural radiation exposure of the human. The International Commission on Radiological Protection (ICRP) recommends that assessment of risk from exposure to radon progeny should be based on epidemiological studies relating excess lung cancer in miners to radon exposure [2]. In annual report of the ICRP for protection against Radon for general public at home there is a recommended conversion coefficient the annual exposure to radon concentration and radon exposure to effective dose. Commission to provide a conversion from radon exposure to effective dose has not used a dosimetric approach, this conversion has been obtained by a direct comparison of the detriment associated with unit effective dose and unit radon exposure. The detriment on exposure to radon progeny of  $1 \text{ mJhm}^{-3}$  is equivalent to effective dose 1,1 mSv for members of the public [3]. In the present investigation, we will be study the dose relevant factors from continual monitoring in real homes into account getting more accurate estimation of  $^{222}\text{Rn}$  the effective dose. The dose relevant parameters include the radon concentration, the equilibrium factor (f), the fraction ( $f_p$ ) of unattached radon decay products and real time occupancy people in home.

## **Materials and Methods**

The ICRP recommends the assessment of risk from exposure to radon progeny is published in the introduction. It is accepted model for estimation effective dose since the

commission had not used a dosimetric approach for radon, this conversion has been obtained by a direct comparison of the detriment associated with a unit effective dose and unit radon exposure. The aim of this study was the estimation the variation of effective doses from the different time occupancy people in room together with analysis continual measurement of the radon concentration, the equilibrium factor ( $f$ ), the fraction ( $fp$ ) of unattached radon decay products.

We used the continual monitors RADIM 2C, WLM-30, FRITRA IV and integral monitor RM-1 for the establishment of the input parameters. The gamma dose rate monitor we used Eberline (FH 40F2). Continuous measurement with time interval 30 min of radon concentration were performed by radon monitor RADIM 2C. Radon progeny, equilibrium factor were measured continuously by WLM-30 with sampling time 30 min. The detectors are ruggedized custom designed silicon barrier diode detector mounted in the base of the filter-detector assembly. For determination of the unattached fraction used Radon and Radon progeny Monitor FRITRA IV. Radon progeny collected on open millipore filter with diameter 16 mm and on filter under mesh. The unattached fraction is calculated from result of the free filter and mesh measurement.

The result of the measurement are time courses of radon concentration, the equilibrium factor ( $f$ ), the fraction ( $fp$ ) of unattached radon decay products and integral value of radon concentration. The measurement were made one week four times per year in different seasons, with different ventilation and time occupancy people indoor home. The aim of this study was the assessment real time occupancy people in rooms from the measurement by registration sensors. We estimated weekly effective dose from the daily doses by using quantification average concentration during the real time occupancy and calculated value for one year. The next result was the time course of the fraction ( $fp$ ) of unattached radon decay products, which is defined as the ratio of the equilibrium equivalent radon concentration in the unattached state and the total equilibrium equivalent radon concentration. The quantitative significance for this estimation could be modified by physical factors not included in the definition such as unattached fraction set apart from dosimetric model where is the one important of the input parameters. We compared the variation of this values during individual measurement.

## **Results and Discussion**

Our measurement was realized in real occupied ground floor house in Beroun, which consisted of three rooms – a kitchen, a living room and a sleeping room. The measurement

Table 1. Average weekly radon concentration measured in different rooms

Rooms	Radon concentration (Bq.m <sup>-3</sup> )				Radon concentration (Bq.m <sup>-3</sup> ) during occupancy people					
	May	Dec	Mar	Aug	May	Dec	Mar	Aug		
Kitchen	536	452	522	307	249	434	447	517	277	218
Living room	746	535	1075	555	224	395	522	1066	536	331
Sleeping room	532	553	899	635	353	279	682	922	617	208

that proceed approximately one week was taken four times, in May and in December 2002 and in March and August 2003. Radon and its daughter products depend on two parameters. These parameters are total volume

entry rate from all sources and ventilation rate, which change during the day. The result of our measurement is the time course of radon concentration, average weekly radon concentration and weekly radon concentration during occupancy people measured in different rooms presented in Table 1. The average values differs in each room. The highest radon concentration was measured in the December and lowest value in August that is caused by temporary variation and dissimilar room ventilation. The lowest fluctuation of the average radon concentration and the average radon concentration during occupancy people was observed in winter, when the ventilation in rooms is minimal. We measured radon concentration two weeks in May and we compared the values in each week. We can see that they differ most in living room, where people spent the most of their time indoor during the day. The lowest variability is in the kitchen, where people spent on average only two hours per day set apart the living room, where they spent five hours and in the sleeping room eight hours mostly between 12 pm and 7 am. We estimated the real time occupancy people in individual rooms by registration sensors and we tried to estimated the year effective doses and compared with assessment from the integral measurement by electrets  $H_E^{(1)}$ (mSv) adequate 7000 hours

Table 2. Estimated year effective doses from inhalation of radon and its daughter products

Rooms	$H_E^{(1)}$ (mSv)				$H_E^{(2)}$ (mSv)				
	May	Dec	Mar	Aug	May	Dec	Mar	Aug	
Kitchen	8,72	11,19	7,79	4,28	0,36	1,19	0,65	0,41	0,06
Living room	10,97	18,79	9,34	4,94	0,88	3,3	2,57	2,65	0,23
Sleeping room	9,51	15,46	10,69	5,56	1,96	5,24	4,15	5,44	0,4
Average	9,73	15,15	9,27	4,93	Total	6,47	7,37	8,5	0,69

Удалено:

indoors and equilibrium factor of 0,4. We counted the year effective doses  $H_E^{(2)}$  (mSv) from individual weekly estimations getting the real time occupancy that changes in rooms and measurements. The results of the estimation the year effective doses are presented in Table.2

We compared the estimations by using the real time an average radon concentration during occupancy and by using integral average radon concentration from the integral measurement and we can see that values are lower than using 7000 hour years year. This value decreased down 14 % because of the fact that people occupied the house only three days and measuring lasted seven days in August. Difference in winter between values is 51 % during seven days occupancy regime that is similar in case May (8,3% ) and March (33,5 %).

According methodology ICRP we used for this estimation average value equilibrium factor of 0,4. The result of our measurement is the time course of the equilibrium factor (f)

Table 3. Measured equilibrium factor f and unattached fraction  $f_p$

Room	f				$f_p$			
	AM	SD	Min	Max	AM	SD	Min	Max
March								
Kitchen	0,48	0,11	0,22	0,71	0,064	0,09	0,0087	0,47
Living room	0,45	0,15	0,06	0,85				
Sleeping room	0,42	0,09	0,1	0,76				
December								
Kitchen	0,41	0,13	0,12	0,76	0,052	0,03	0,018	0,154

and of the fraction ( $f_p$ ) of unattached radon decay products. The arithmetic mean (AM), standard deviation (SD), minimum and maximum of these parameters are presented in Table 3. The average values changed in different rooms and in time and show large

variability. Parameter of the equilibrium factor (f) is different in some cases than value 0,4 that is recommended by ICRP.

## Conclusion

The result of the measurement are the time courses of radon concentration that are based on estimation effective doses together with assessment of the real time occupancy people indoor. We found out by analysis that year effective dose is lower than effective dose estimated by ICRP recommendation from the integral measurement that included only average radon concentration. Our analysis of estimation effective doses using measurement of several physical parameters was made only in one case and for the better specification is important to measure in different real occupancy houses.

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# RETROSPECTIVE CHROMOSOME ABERRATION ANALYSIS OF FORMER URANIUM MINERS

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## **Introduction**

Despite of the large occupational population worldwide of underground uranium miners rather few studies have been published on the cytogenetic status or injuries of the exposed persons (Brandom et al 1978, Köteles et al 1988). In this paper we present our data collected in the period of 1981-1985 on 165 persons exposed by different radon concentrations expressed in working level month (WLM) units from 100 up to 600. Following the decommissioning of the uranium mine in Hungary in 1997 cytogenetic status of 131 persons were within a follow-up-study of their health conditions initiated by the Hungarian Academy of Science. The persons have terminated their underground activities 5 to 20 years before testing. The comparison of the two datasets suggest a long-term persistence of cytogenetic alterations above the population average values in large percentages of persons investigated.

## **Materials and methods**

Whole blood samples were taken in heparinized tubes and cultured for 48 hours in the presence of bromodeoxyuridine. The chromosome sets were scored following FPG staining from duplicate slides. Only cells within the first cell cycle were evaluated at least 100 metaphases from each person. The persons investigated were categorized into 6 groups according to their cumulative exposures from 100 up to 600 WLM. The number of persons and the number of cells scored in each group are listed on Table 1.

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**Table 1. Number of uranium miners tested and the number of lymphocytes scored in the various exposure groups.**

Exposure category	No of persons	No of cells scored	No of persons and cells years after underground work					
			3-7 yrs		8-12 yrs		18-25 yrs	
			person	cell	person	cell	person	cell
I	27	2554	4	559	22	4080	4	632
II	32	2720	2	300	38	6189	27	3746
III	32	2814			12	1968	10	1720
IV	25	1866			6	1080		
V	21	1880			4	800	4	520
VI	28	2173			4	800	4	520
Total	165	14007	6	859	86	14917	49	7138

## Results and Discussion

*Data obtained between 1981-1985.*

The frequency of chromosome aberrations in peripheral lymphocytes of exposed persons in function of their exposures in WLM are shown on Table 2. It is seen that the aberration frequencies are higher than the values in unexposed population in all exposure groups and for all types of aberrations scored. Among the various exposure groups no considerable dependencies on WLM categories were found in respect of the total number and chromatid type aberrations (deletions). The frequency of chromosome-type aberrations (dicentric) suggests an increasing tendency as the cumulative exposure gets higher, especially in the IV and V categories the frequencies reach a plateau, then either they level at this value or decrease. This phenomenon was observed by other authors, too (Brandom et al 1978). The persons tested have been worked for years in underground mines. In the meantime the working conditions have been improved considerably, therefore, their exposures were decreasing. Therefore an analysis was also performed, when the dicentric aberration frequencies were related not to the cumulative exposure, but to the average yearly exposure. It was evident that the frequency of dicentric was increasing considerably.

*Data obtained between 1998 and 2002.*

The dataset on the frequency values obtained 5 to 20 years after finishing the underground work is seen on Table 3. It was observable that the dicentric aberrations decreased while the deletions were found almost unchanged during this period. The frequency of the total aberrations was decreased, too.

Large fraction of persons carrying aberrations as well as the long-term persistence of aberrations suggest that certain clastogenic factors might maintain the cytogenetic alterations similarly to the clean-up-workers in Chernobyl (Emerit et al 1994, Emerit 1998). It has to be mentioned, however, that long-acting chromosome aberrations were demonstrated earlier in A-bomb survivors (Awa 1983), Japanese fishermen exposed to fallout radiation (Ishihara and Kumatori 1983), therapeutically irradiated ankylosing spondylitis patients (Buckton 1983). Further investigations are planned to analyse a few response-modifier factors like the tumor necrosis alfa (TNF $\alpha$ ) as well as the total antioxidant status of former miners.

**Table 2. Frequency of chromosomal aberrations in the lymphocytes of active uranium miners in various exposure categories.**

Exposure category	No of scored cells	Frequency of aberrant cells	Dicentrics	Acentrics	Deletions	Total aberrations
WLM		$\times 10^{-2}$	$\times 10^{-3}$	$\times 10^{-2}$	$\times 10^{-2}$	$\times 10^{-2}$
I	2554	4,07	4,9	3,73	1,58	5,88
II	2720	5,40	5,5	3,97	1,51	6,14
III	2814	5,84	5,6	4,21	1,98	6,79
IV	1866	6,22	8,1	4,45	2,20	7,56
V	1880	5,02	8,4	3,27	1,58	5,76
VI	2173	6,26	8,5	4,12	1,47	6,54
Control			0,5-1	1-3		1-3

**Table 3. Changes of the cytogenetic parameters years after finishing underground work in former uranium miners.**

Exposure category	Time	Dicentrics	Acentrics	Deletions	Total
		aberrations			
	Year	$10^{-3}$	$10^{-2}$	$10^{-2}$	$10^{-2}$
I.	in active years	4,9	3,73	1,58	5,88
	after 3-7 years	5,4	1,25	2,15	3,94
	after 8-12 years	4,2	1,57	1,79	3,77
	after 18-25 years	1,6	1,90	1,27	3,36
II.	in active years	5,5	3,97	1,51	6,14
	after 8-12 years	4,7	1,53	1,66	3,62
	after 18-25 years	4,0	1,10	1,74	3,20
III.	in active years	5,6	4,21	1,98	6,79
	after 8-12 years	3,0	1,93	2,03	4,27
	after 14-23 years	2,9	1,16	1,45	6,91
IV.	in active years	8,1	4,45	2,20	6,22
	after 8-12 years	1,9	1,39	1,94	3,52
V.	in active years	8,4	3,27	1,58	5,76
	after 8-12 years	2,0	1,00	0,50	1,75
	after 17-23 years	1,35	7,70	2,50	4,62
VI.	in active years	8,5	4,12	1,47	6,54
	after 8-12 years	2,5	6,30	1,13	2,25

In summary, the frequency of chromosome aberrations of uranium miners was found increased in function of their exposure to radon. The comparison of the miner's categories 20 years ago and in the recent years demonstrated the long-term existence of aberrations for many years after completion of underground mining activities.

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# LEUKAEMIA AMONG URANIUM MINERS – LATE EFFECTS OF EXPOSURE TO URANIUM DUST

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## **Introduction**

The evidence of the risk of leukaemia from ionizing radiation is mainly based on studies of atomic bomb survivors and populations exposed to x rays for diagnostic or therapeutic purposes. There are also an increasing number of occupational studies. The evidence of leukaemia risk from radon among uranium miners is limited. The presentation reports on recent observations among Czech uranium miners by 1999.

## **Methods**

The cohort study of uranium miners was established in 1970 in order to analyze the risk of lung cancer from radon (Ševc et al, 1971). The details of the studies are given elsewhere. In brief, the cohort involves two groups of uranium miners exposed in two different periods. A total of 4339 miners from the older cohort (S) were employed in the Jáchymov region (West Bohemia) in the period 1948-63 (Tomášek et al 1994). The other 5621 miners belong to the newer cohort (N) and were employed in the Příbram mines (Central Bohemia) during 1968-88 (Plaček et al 1997).

Analyses are based on the relative risk (RR) model with a linear dependence on cumulated exposure (W) in the form  $RR = 1 + b W$ . Alternatively, the relative risk is analyzed in dependence to duration of exposure. Analyses related to leukaemia subtypes are based on simple relative risk model based on hewer equivalent duration in two categories: less than 10 years and 10 years and more.

## **Results**

A total of 9960 workers have been followed to the end of 1999 representing 267 thousands person-years. With the exclusion of lung cancer, there were practically no other

significant malignancies in this study related to radon exposure. In later periods of follow-up, however, some increase of haematopoietic cancers was observed, including 29 cases of leukaemia, mostly myeloid (16) and chronic lymphatic (7).

The ratio of the observed number to the nationally expected number is 1.31, 90%CI: 0.88-1.88. When the risk is related to cumulated exposure to radon progeny, no statistical association is seen ( $p=0.172$ ), but the trend with duration of underground work is significant ( $p=0.015$ ). These contrasting results suggest that the risk of leukaemia is probably related to other noxae present in the underground environment. In comparison to estimated doses to the lungs, which are based on extensive radon measurements commencing already in 1949, doses to the red bone marrow can only be estimated with some uncertainty. One such estimate was suggested by Jacobi (1995) in a report on risk of extra-pulmonary cancers from radiation exposure among workers in German uranium mining company. Extrapolating this indirect dosimetric approach and results of measurements of alpha activities in the dust and external gamma exposure in Czech uranium mines conducted since the 1970s, annual doses to the red bone marrow can be estimated as follows:

Table 1: Annual estimated equivalent doses to the red bone marrow

	1950-59	1960-69	1970-89
drilling	34 mSv	23 mSv	12 mSv
50% drilling	22 mSv	15 mSv	8 mSv
other activities	10 mSv	7 mSv	4 mSv

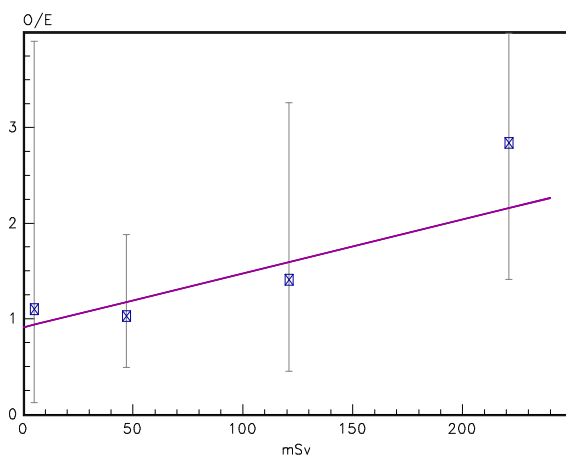


Figure 1: Relative risk of leukaemia (O/E) in dependence to estimated dose

The risk coefficient (excess relative risk per sievert) corresponding to these estimates in the present study of Czech miners is  $ERR/Sv=7.1$ , 90%CI: 3.6 – 13.9 (Figure 1). The dose estimates, however, are subject to a considerable uncertainty depending among others on dust

content, ore mineralization, or type of drilling. Therefore, it is perhaps more practical to assess the risk using only duration of exposure that is modified according to job category. As the red bone marrow estimates for hewers are about two-fold, the modification can consist in accounting only half of the year for job categories other than hewer (Table 2).

Table 2: Relative risk of leukaemia (O/E) in relation to modified duration of exposure

exposure duration <sup>a</sup>	PY	O	O/E
0-4 y	179 503	10	0.79
5-9 y	64 656	10	1.45
10-14 y	19 152	7	3.42
15-34 y	4 134	2	3.82
total	267 445	29	1.31

<sup>a</sup> non-hewer jobs are accounted as 1/2 year

Using this approach, the ERR per year of the hewer's work is 0.15 (90%CI: 0.08 – 0.27). According to this model, the risk of leukaemia is doubled after 6.5 years of the hewer's work. Risk for leukaemia subtypes can only be analyzed in a simple dichotomic design (Table 3).

Table 3: Leukaemia subtypes in dependence to modified duration of exposure

	O	O/E	90%CI	<10y	>10y	RR*	90%CI
nonCLL	22	1.27	0.85-1.83	16	6	3.06	1.56 - 5.98
CLL **	7	3.89	1.82-7.30	4	3	4.93	1.91 - 12.8
all types	29	1.46	1.04-2.00	20	9	3.50	2.02 - 6.06

\*risk relative to modified duration of exposure <10years

\*\*chronic lymphatic leukaemia

## Discussion

Results on leukaemia risk among uranium miners are scarce. In a report from 11 miner studies (Darby et al, 1995), no association to radon exposure (based on 69 cases) was observed. However, the results of follow-up by 1990 among Czech uranium miners suggest some increase with duration of exposure (Tomášek et al, 1993). Generally, the evaluation of leukaemia risk is limited by low numbers of cases. In the male population, leukaemia incidence is lower by one order of magnitude in comparison to lung cancer and is manifested mainly in age groups over 55. Therefore, more evidence can be expected in long follow-up or by combining several studies.

In radiation induced leukaemia, chronic lymphatic leukaemia (CLL) is usually excluded, as in most studies no radiogenic association was observed for this subtype. In the present study all subtypes were combined. However, the association in the simple dichotomic model was stronger for CLL than for other subtypes combined, although not significantly.

Although the assessment of the leukaemia risk is based on an approximate estimation of dose, the estimated value of excess relative risk per Sv is consistent with findings of the LSS study, where the risk coefficient in linear model below 4Sv is 8.0 per 1 Sv. However, the evaluation of occupational exposure should be based on more objective data, eg. duration of exposure. As there are differences in dose estimates for drilling and other activities, duration of exposure should be modified according to the job. In the absence of detailed data, the estimation of 50% for other than heavier activities is a simplification.

## Conclusions

Results of leukaemia study in a cohort of nearly 10 000 uranium miners based on 29 observed cases show significant association to chronic exposure. The largest contribution is from the uranium dust. Although there are estimates to the red bone marrow, it seems to be more practical to use a model based on modified duration of exposure, where non-heavier exposure are accounted by 50%. The evaluation of leukaemia subtypes is limited by low numbers. More accurate results can be expected in an extended follow-up or by pooling several studies.

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# **Occupational Diseases in Uranium and Ore Miners in Connection with Radiation Exposure in the Czech Republic**

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## **Abstract**

Dozens cases of diseases are submitted to judgement as occupational diseases every year in the Czech Republic. Patients or attending physicians suggest that these cases are caused by occupational ionizing radiation. Only a part of these cases is qualified as occupational disease. The term “occupational disease” is rather a juridical term which underlies the right to financial compensation. The causal association with exposure to ionizing radiation cannot be indisputably verified by expert medical opinion.

Most diseases, which are proposed as occupational disease, are malignant tumors of the lungs. Total majority of judged cases are lung cancers from radioactive agents. In 2002, a total of 33 cases of lung cancer in former uranium or ore miners have been acknowledged as occupational diseases (1). The decision about occupational disease is derived from probabilistic approach based on estimation of probability of causation of irradiation on disease origin (methodical guideline No. 15 of Ministry of Health Bulletin, part 9, 1998).

The presented paper gives a general information about all judged causes of occupational diseases in former uranium and ore miners in the Czech Republic in 2002.

## **Introduction**

A total of 35 occupational diseases caused by occupational ionizing radiation have been acknowledged in uranium and ore miners in the Czech Republic in 2002 (1). Most of acknowledged diseases – a total of 33 cases – were lung cancers from radioactive agents (Item 6 Chapter III of the List of Occupational Diseases of Government Regulation No. 290/1995). The other two cases were diagnosed as acute and chronic myeloid leukemia (Item 1 Chapter II of the List of Occupational Diseases).

Paper gives a general information about all judged cases of occupational diseases in uranium and ore miners in the Czech Republic in 2002. There were a total of 72 cases submitted for judgement, 67 lung cancers, 1 acute myeloid leukemia, 1 chronic myeloid leukemia, 1 basaliom, 1 cancer of larynx and 1 cancer of nasal septum.

## **Methods**

It should be stressed that the term “**occupational disease**” is at first juridical term which underlies the right to financial compensation. Nevertheless the causal association with exposure to ionizing radiation cannot be always indisputably verified by expert medical opinion.

Acknowledgement of occupational disease is procedure which must correspond to requirements of **Regulation of Ministry of Health No. 342/1997**. The acknowledgement of

occupational diseases is executed in the medical centers of occupational diseases (§1, §3). Procedure of compensation is initiated by written report about suspicion of occupational disease. The report is submitted to competent regional medical center of occupational diseases. The procedure can be initiated by the clinician or by the patient (§2 section 1). The physician of the State Office for Nuclear Safety judges conditions of origin of disease which can be caused by work with ionizing radiation or radioactive agents (§2 section 3 b).

Occupational diseases are defined as “acute intoxications caused by negative effect of chemical agents on health, diseases caused by negative effect of chemical, physical, biological or another negative effects if they arose from conditions” presented in the **List of Occupational Diseases, Annex of Government Direction No. 290/1995**. Following three chapters of the list relate to effect of ionizing radiation:

**Chapter II, item 1** – disease caused by ionizing radiation

**Chapter III, item 6** – lung cancer from radioactive agents

**Chapter IV, item 1** – skin diseases caused by physical, chemical or biological factors

Total majority of diseases in uranium and ore miners submitted for compensation as occupational disease caused by ionizing radiation are malignant tumors. The decision about occupational disease is derived from probabilistic approach based on knowledge of relation between dose and effect on target tissue where tumor arose. The **criterion of prevailing probability** is valid, i.e. causality is accepted if the probability of causation of irradiation on tumor origin is greater than probability of spontaneous tumor incidence.

The procedure recommended for judgment of cases of lung cancer from radioactive agents (Chapter III, item 6 of the List of Occupational Diseases) is based on this principle and it is published as **Methodical Guideline No. 15 of Ministry of Health Bulletin, Part 9, 1998**. This methodical guideline is designated for the assessment of lung cancer cases among workers or former workers in uranium mines or another workplaces with considerably high concentration of radon and its short-lived progeny. The disease must be proved or there must be reasoned suspicion of it. The term “reasoned suspicion” is understood as the situation when the diagnosis of lung cancer is considered by clinician as vindicated (with no histologic or bronchoscopic verification, too).

The decision about occupational disease is based on the estimation of **probability of causation** of irradiation on disease origin (PC).

$$PC = R_p / (1 + R_p),$$

where  $R_p$  is the excess relative risk of lung cancer in defined level of irradiation and in defined exposure periods (table No. 1).

$$R_p = \sum k_i W_i,$$

where  $k_i$  are risk coefficients depending on time-since-exposure and age-at-exposure and  $W_i$  are corresponding exposures.

The risk coefficients are derived from recent results of epidemiological studies in uranium miners (2). They are more precise and more favorable to the patient. The coefficients are valid for lung cancer in all histologic forms. The exposures can be found on individual dosimetric cards. The employer must keep the cards for all workers. **If PC value is higher than 0.5 the probability of causation is assessed as prevailing.** The case is qualified to financial

compensation. If PC value is **in border zone 0.4 – 0.5** the compensation is possible with respect to other circumstances. If the PC value is **lower than 0.4** the probability of causation is assessed as low and the case is not qualified to compensation.

The assessment of myeloid leukemia cases in two former uranium miners was also based on the estimation of probability of causation of irradiation on disease origin. It was necessary to estimate the dose to bone marrow from external irradiation and from inhalation exposure to radon and its progeny and from inhalation of powder particles containing long-term alpha emitters. The dose received from inhalation of uranium powder is the most important (80%). The dose estimation was based on the results of Czech and German studies (3, 4, 5). The doses in miners working in Czech uranium mines after 1969 can be estimated on 12 mSv/year in quarrymen and 4 mSv/year in other professions (table 2).

**Table No. 1:**  
**Lung cancer**  
**Relative risk coefficients per WLM**  
**(dependence on age at first exposure and time since exposure)**

Age at first exposure	Time since exposure			
	0 – 4	5- 14	15 - 24	25 +
- 29 years	0	0,205	0,084	0,038
30 – 39 years	0	0,029	0,027	0,02
39 - years	0	0,009	0,008	0,004

$$PPS = R_p / (R_0 + R_p)$$

$R_0$  .... 100%

$$R_p = \sum_i k_i \cdot WLM_i$$

$k$ .... relative risk coefficient

PPS < 0,4

Probability low

PPS 0,4 – 0,5

Probability borderline

PPS > 0,5

Probability high

The value of excess relative risk coefficient was 7,9 per 1 Sv. Excess relative risk was significant and PC values were higher than 0,5 in both cases. The probability of causation was assessed as prevailing and both cases of myeloid leukemia were qualified to compensation.

**Table No. 2:**

**Leukemia**

**Estimate of dose to bone marrow per year in uranium miners**

<b>Profession</b>	<b>1950-1959</b>	<b>1960-1969</b>	<b>1970-</b>
<b>Quarryman</b>	<b>34 mSv</b>	<b>23 mSv</b>	<b>12 mSv</b>
<b>Other</b>	<b>10 mSv</b>	<b>7 mSv</b>	<b>4 mSv</b>

**ERR/Sv = 7,9**

**Conclusion**

A total of 72 cases were submitted to judgement of conditions of disease origin to the National Radiation Protection Institute in 2002. 67 cases were lung cancers, 1 case was chronic myeloid leukemia, acute myeloid leukemia, basaliom, cancer of larynx and cancer of nasal septum.

The probability of causation was assessed as prevailing in 32 cases of lung cancer, borderline in 5 cases and low in other 30 cases of lung cancer.

The probability of causation was prevailing in both cases of myeloid leukemia.

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# The Health Status of the Population Neighbouring the Nuclear Power Plants in Slovakia

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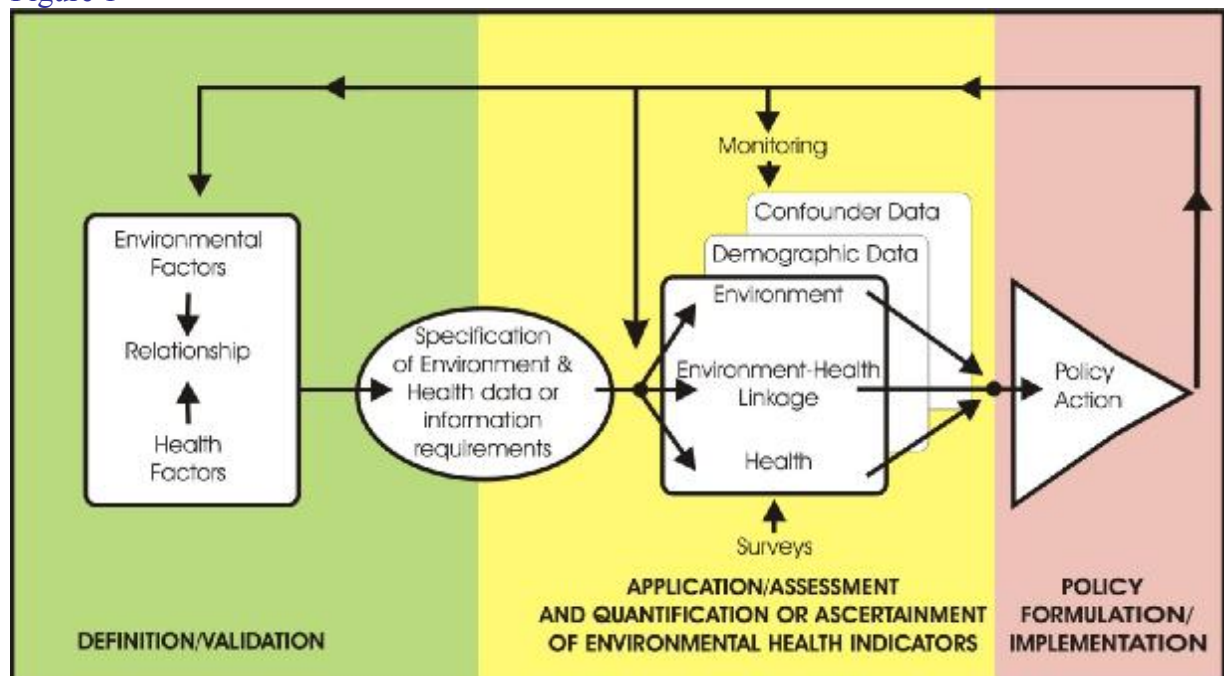
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## Introduction

Nuclear power plants are regarded as potential sources of low doses of ionising radiation. We therefore consider it important to assort and evaluate health and environmental analyses focused on their influence, so that we can react promptly in case of necessity.

The whole process of the monitoring and investigation of the health condition of the population, as well as the resulting decisions, is characterised best by the following scheme – carefully applied and checked continuously as to its proper functioning.

Figure 1



(D.Briggs, C.Corvalán, M.Nurminen: Linkage methods for environment and health analysis, WHO, Geneva, 1996)

## Study Aim

Our objective is to ascertain the actual state of the indicators of health in individual villages within the area under investigation, to ascertain the trend of the development of indicators in the area surrounding the nuclear power plant included in the study, to find whether the occurrence of an indicator is accidental or whether it is determined in the village, make a comparison with another area and with the situation in the Slovak Republic as a whole and, consequently, to determine possible influence of the Power Plant on the indicator's value.

### *We follow several lines*

we ascertain which health indicators can be gathered on desired level of accuracy and reliability

we select those that are relevant and calculate them in order to be able to make comparisons in all the villages of the Slovak Republic

we determine the health risk of the power plant as a point source by special mathematic methods from selected and calculated indicators and determine if there is a correlation between the rate of an indicator and the operation of the nuclear power plant.

### Health Indicators

We use methods applied in the field of population epidemiology and focus our interest on the health condition of the selected population as a whole, described in its entirety by the complete range of health indicators. We do not follow the fate of any concrete individuals or any specific diseases.

Used are indicators calculated from basic demographic calculation operations and national registries of diseases.

Selection of indicators and their calculation is performed in accordance with the recommendations of the World Health Organization (WHO) in Geneva.

The complete set of indicators according to WHO being quite extensive, a selection was made of those followed in the official national statistics and therefore warranted by the state. A further narrowing of the selection was made taking into account possible impact of the power plant's operation on human health.

The health indicators were calculated for each village in Slovakia individually and they were evaluated separately for each village in the territory of a power plant.

The population in power plants' neighbourhood and, for the sake of comparison, also the entire population of Slovakia, are evaluated by means of health indicators.

### *Criteria for Environmental Health Indicators :*

followed on the entire territory of Slovakia, to enable territorial comparisons

followed long-term and continuously, if possible without any significant interruptions

collected by the same, internationally accepted methodology

verified by the state authority

and, of course, still valid

A total of 43 individual health indicators were followed which characterise the demographic composition of the population in a village, reproductive health, morbidity due to selected diseases, total and premature mortality, as well as mortality caused by selected diseases.

For the purpose of mathematical analyses the following 15 indicators are used:

**Table 1 Selected indicators analysed by fuzzy c-cluster analysis**

Indicator	Contents	Selection reason
1	<i>PYLL100</i> Potential Years of Life Lost (PYLL) per 100.000 Inhabitants	Premature Mortality
2	<i>PYLL1</i> Potential Years of Life Lost (PYLL) per one Death	Premature Mortality
3	<i>REC00</i> Number of total Cancer Deaths per 100.000 Inhabitants (classified as C00 – C99 in the ICD, ver.10)	Selected Causes of Death
4	<i>REC1526</i> Number of Alimentary Canal Cancer Deaths per 100.000 Inhabitants (classified as C15 – C26 in the ICD, ver.10)	Selected Causes of Death
5	<i>REC9195</i> Number of Leukaemias Deaths per 100.000 Inhabitants (classified as C91 – C95 in the ICD, ver.10)	Selected Causes of Death
6	<i>REC34</i> Number of Lung Cancer Deaths per 100.000 Inhabitants (classified as C34 in the ICD, ver.10)	Selected Causes of Death
7	<i>REI00</i> Number of Cardiovascular System Deaths per 100.000 Inhabitants (classified as I00 – I99 in the ICD, ver.10)	Selected Causes of Death
8	<i>HUS</i> Total Mortality (Number of Deaths per 1000 Inhabitants)	Basic Indicator of the Mortality
9	<i>HUM</i> Total Male Mortality (Number of Male Deaths per 1000 Men)	Basic Indicator of the Mortality
10	<i>HUZ</i> Total Female mortality (Number of female Deaths per 1000 Women)	Basic Indicator of the Mortality
11	<i>PUS</i> Proportion of Premature Deaths (before the Age of 65 Years)	Basic Indicator of the Premature Mortality
12	<i>PUM</i> Proportion of Premature Male Deaths (before the Age of 65 Years)	Basic Indicator of the Premature Mortality
13	<i>PUZ</i> Proportion of Premature Female Deaths (before the Age of 65 Years)	Basic Indicator of the Premature Mortality
14	<i>PPOD</i> Proportion of liveborn Children with low Birth Weight under 2500 Grams	Reproduction Malformation
15	<i>SPOTPOT</i> Proportion of spontaneous Abortions per total Conception	Reproduction Malformation

We will analyse in detail some of them as a sample and then draw a conclusion from all.

### ***Data***

Data necessary for the calculation of indicators are obtained from the following sources:

The complete Register of Deaths, The Register of Births, and The Register of Abortions in Slovakia in 1993, 1994, 1995, 1996, 1997, 1998, 1999, 2000 ( State Statistics of the Slovak Republic)

The complete balance of the villages, age structure of the villages in 1993, 1994, 1995, 1996, 1997, 1998, 1999, 2000 ( State Supernumerary of the Slovak Republic)

National register of Chronic Lung Diseases and the national register of Tuberculosis ( The National Institute of Tuberculosis and Respiratory Diseases)

The land register of the villages according to map letters in ARC VIEW dated 1996

Structure of pollution from the nuclear power plant Bohunice and the nuclear power plant Mochovce for the years followed.

We examined the health of the population living in the vicinity of nuclear power plants in an area specified in the protection area guidelines, in the case of the nuclear power plant Bohunice within the radius of 30 km and in the case of the Nuclear Power Plant Mochovce within the 20 km radius.

Regular reports on the health condition of the population are prepared since 1992.

### ***Methods***

We use methods applied in population epidemiology. The entire evaluated population is described by global health indicators as a whole, or in parts.

We follow no concrete illnesses or deaths.

Our analysis is based on modern multi-dimensional mathematical and statistical methods, such as fuzzy cluster analysis or spatial statistics. As a result we always get accurate mathematical specification of the level of the calculation's importance ( statistical level of significance).

Health risks are evaluated by several mathematical methods. Similarity of the indicator's value in connection to the point of the emission's source is evaluated with the assistance of

fuzzy cluster analysis. Similarity of the value and development of indicator in a village without respect to the source is evaluated by means of fuzzy C cluster analysis of the time periods. Randomness or determination of the indicator's value in a village is ascertained by methods of spatial autocorrelation. Individual year- indicators are added up to the short time periods to eliminate the age effect. Time periods also minimize the risks associated with working with small numbers that may affect the results when evaluations are made of conditions in individual small villages.

We therefore speak with mathematical accuracy whether the condition and development of a health indicator in a village are similar to those in the entire examined area, or whether they are different.

We also conclude with mathematical accuracy whether the value of indicator in a village is incidental, or whether there is some causal relationship either with events in the village, or with the wider situation that exists regardless of the boundaries of an individual village ( if similar phenomena are found to occur in clusters)

Only after we find out that a cluster of villages can be confirmed in which the indicator's value is worse for no incidental reason, but rather where there is an apparent connection with the wider surroundings, we start to investigate the possible cause, which, eventually, might turn out to be also the nuclear power plant.

## ***Results***

Our first conclusion is that both nuclear power plants are located in areas where most of the villages are small. The demographic structure of small villages is characteristic and it markedly influences the value of all health indicators. The population in small villages is usually older. It is therefore obvious that rural communities have higher morbidity and mortality than larger towns and cities. Higher long-term values of some indicators in comparison to those for Slovakia as a whole are caused by the age structure of the population and are comparable with the villages with a similar composition of the population, or even better.

## *Model Analysis of the Health Indicators*

### **The Nuclear Power Plant BOHUNICE**

The Bohunice power plant has been in operation for more than 20 years. We have no data that would give a clear picture of the initial state of health indicators. All the indicators are calculated from data recorded during the long, smooth operation of the nuclear power plant. Thus, all the values are "influenced", if such a condition exists.

#### *Selected Indicator of Reproductive Health :*

##### *Proportion of liveborn Children with low Birth Weight under 2500 Grams*

*( so called miscarried Children)*

It is assumed, that this indicator simply shows whether any particular cause exists, resulting in confirmable damage to children's development in its most sensitive stage. It partially compensates the information on the number of children with inborn defects. The value of such a statement would be higher, but it is unavailable for the whole Slovakia.

The period from 1993 to 2000 is investigated, divided into 6 short, partially overlapping time periods.

We evaluated the results by fuzzy c-cluster analysis where we look up villages with similar state and development.

We compared the state and development of indicator in individual villages by means of this analysis, regardless of the number of inhabitants. The result is not distorted by small numbers.

The result of the analysis is a table containing the average values of clusters, a graph of all clusters' trends and a map of villages coloured in accordance with the particular cluster in which they belong . By another method of space autocorrelation we continue investigating whether the state and trend in a village are accidental, or dependent on other circumstances in a village, or wider surroundings.

#### *Long- term Evaluation of the Percentage of Prematurely Born Children*

In the whole of Slovakia the share of prematurely born children varies from 6.6% - 6.8%

The trend shows a slight increase. Villages neighbouring the nuclear power plant Bohunice

are dividable into two limited clusters and trends. The values for the first cluster of villages (dark-blue line in the graph and colour on the map) are well below the national average and there is a trend towards their further decrease. These villages are located in a random pattern around the nuclear power plant Bohunice.

The dark green line in the graph and the same colour on the map indicate villages where the trend and state are similar to those in the whole of Slovakia. Villages not belonging exclusively to any particular cluster are depicted in light colours and they "rather belong to" one of the clusters according to the cluster's colour.

The coefficient of autocorrelation shows, that the situation in the entire area around the nuclear power plant Bohunice is of accidental nature. No clusters of unfavourable values of indicator have been found.

### Conclusion

A long-term favourable situation has been confirmed in the incidence of premature births in the investigated surroundings of the nuclear power plant Bohunice. The value of this indicator is accidental in all the villages investigated and the area as whole is below the average rate of premature births in the whole of Slovakia.

The operation of the nuclear power plant does not influence the reproductive health specified by this indicator.

Figure 2

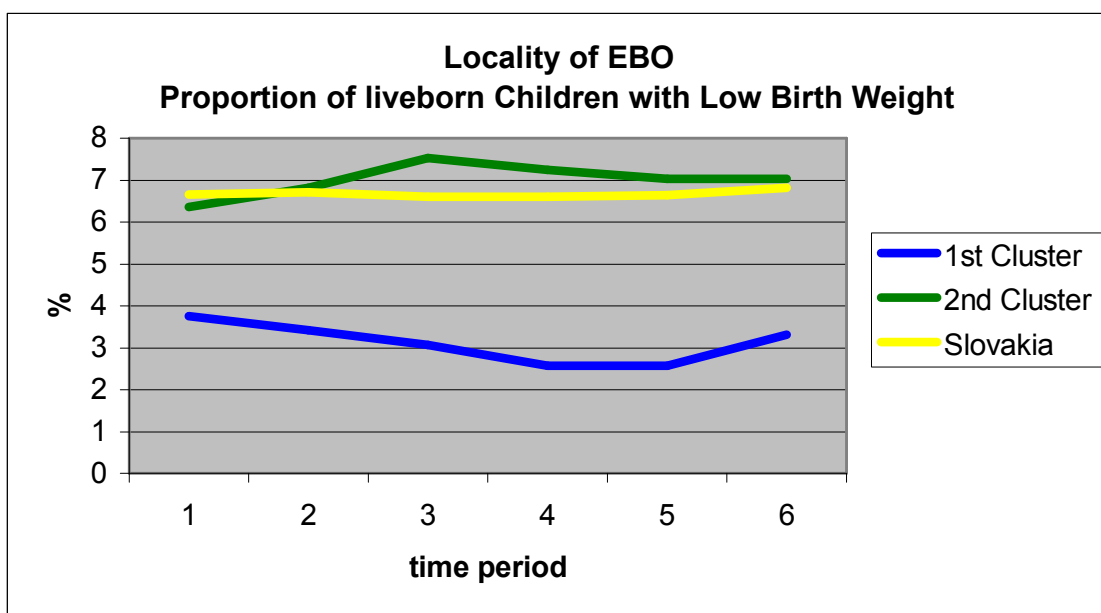
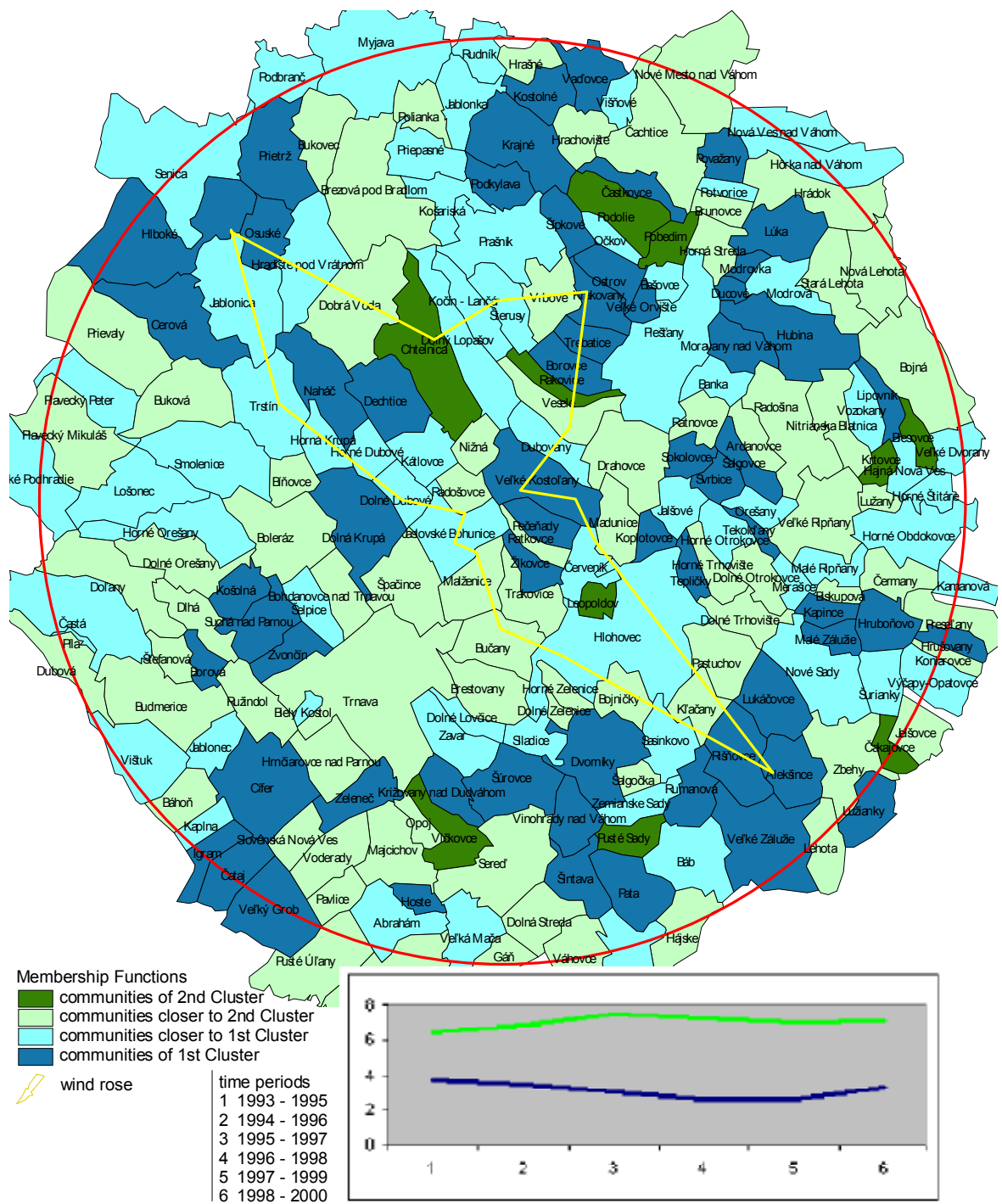


Figure 3

Proportion of liveborn Children with low Birth Weight under 2500 Grams

Locality of NPP Bohunice + 30 km

time periods of fuzzy c-cluster analysis

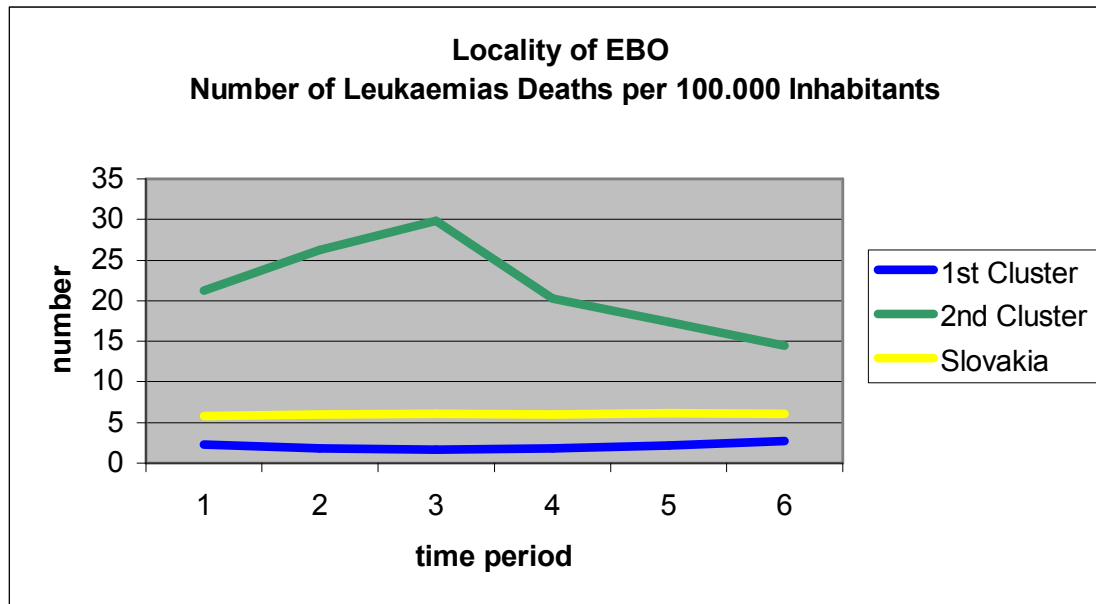




*Selected Indicator of Mortality: Proportion of Leukaemia's Deaths*

Incidence of Leukaemia Deaths is evaluated similarly.

Figure 4



*Conclusion from the evaluation of the proportion of leukaemia deaths in the surroundings of the nuclear power plant Bohunice*

The situation and development of the proportion of leukaemia deaths in the investigated surroundings of the nuclear power plant Bohunice has been found to be unaffected by the power plant's operation and the results of the investigation enable to conclude that the trends found are below those for the whole of Slovakia.

Two villages (Šúrovce and Bojná) appear to be an exception, but both of these rural communities are quite far from the nuclear power plant Bohunice and are located outside the area of potential risk determined by the prevailing winds, or water recipients of the Power Plant.

The value of this indicator is provably accidental in all of the investigated area and no clusters of villages with unfavourable situation were found.

Events in or outside the villages investigated have no provable influence on the indicator's value.

This finding is valid for all the villages within 30 km of the nuclear power plant Bohunice.

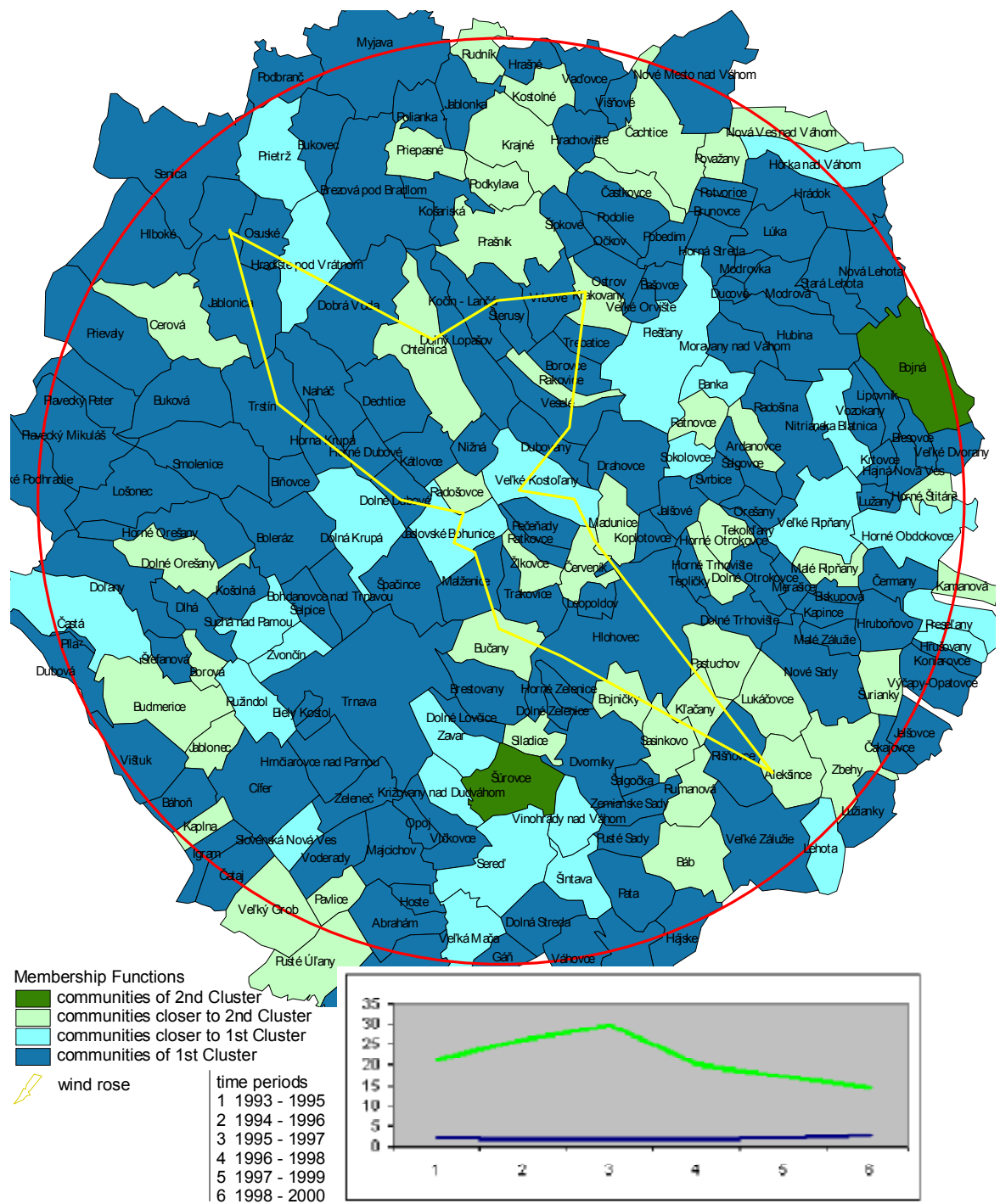
The operation of the nuclear power plant does not influence the rate of leukaemia deaths in the power plant's surroundings.

Figure 5

### Number of Leukaemias Deaths per 100.000 Inhabitants

Locality of NPP Bohunice + 30 km

time periods of fuzzy c-cluster analysis



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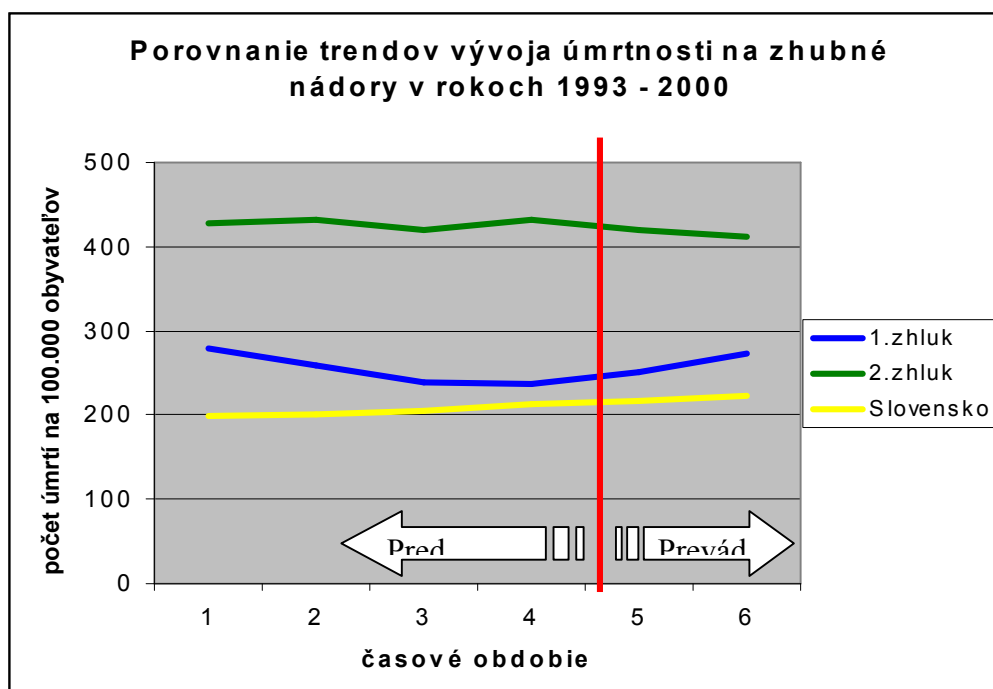
## The Nuclear Power Plant MOCHOVCE

The Mochovce nuclear power plant has been in operation for four years. We recorded the initial values of indicators and included them in the "Preoperation Safety Report". They serve as a comparative value in all the subsequent investigations of the influence of the Nuclear Power Plant's operation on the surrounding area. The values of indicators were calculated on the basis of data which had been recorded before the nuclear power plant was put in operation as well as during its operation (1998 - 2000). It is clear from the above stated, that the values of all indicators can be described as influenced but they should have become visibly "influenced" in the period investigated, if such a change really occurred.

***Selected Indicator of Mortality: Relative Number of total Cancer Deaths (per 100.000 Inhabitants)***

Evaluation of the situation and trends in villages by means of fuzzy cluster analysis and spatial autocorrelation

Figure 6



Cancer-caused mortality in the Mochovce region has been high during the entire investigation period regardless of the operation of the local nuclear power plant. Incidence of cancer in the region had been high long before the power plant was put into operation. Cancer cases increase in number also nationally. Our analysis has divided the villages in the Mochovce area into two distinct clusters, with high and very high cancer incidence - in both of the cases higher than the national average. Both the states are equally divided in all the surroundings of the Nuclear Power Plant Mochovce.

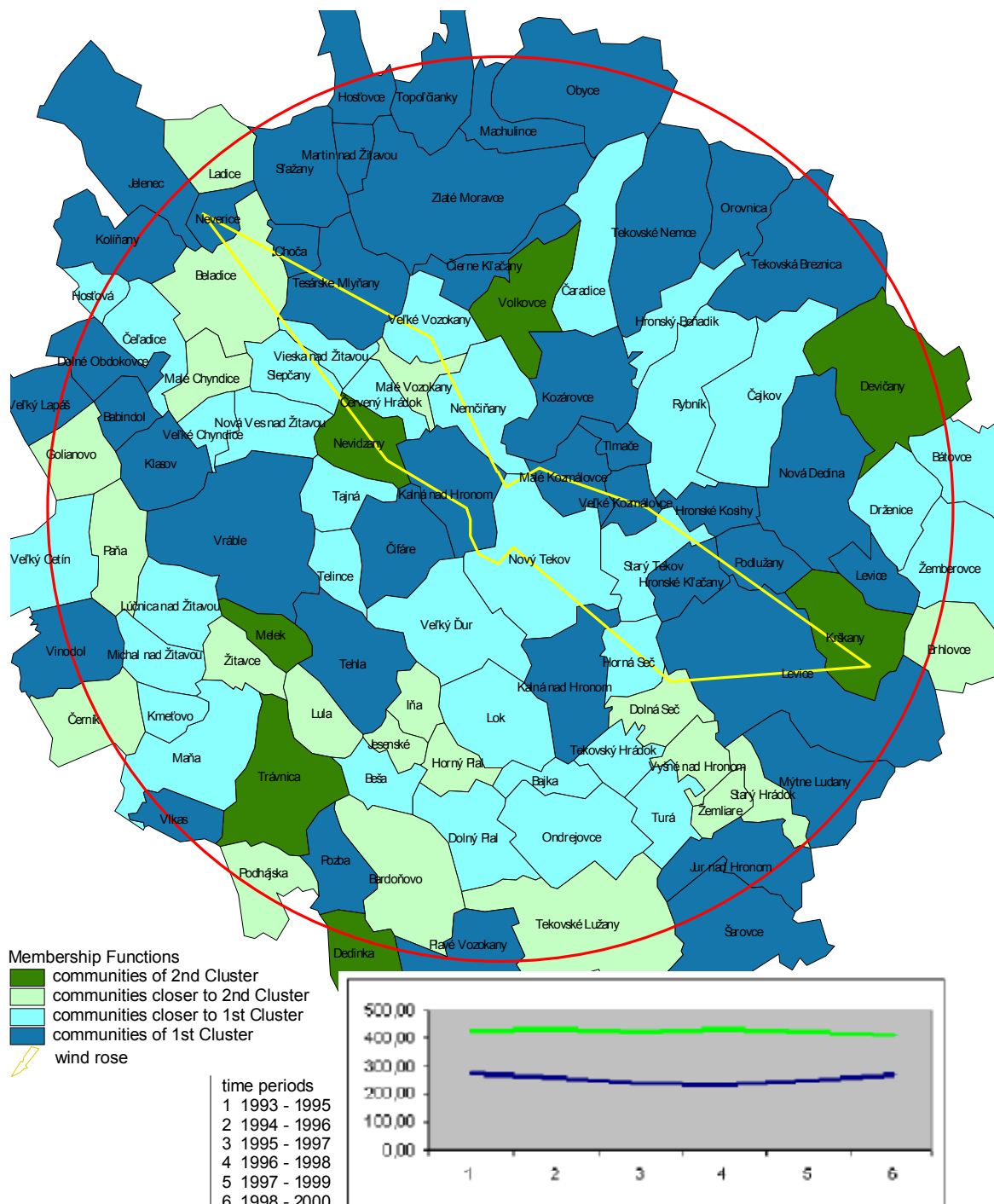
The value of this indicator is provably accidental in the entire investigated area and no clusters of villages with unfavourable situation were found. The trend was not influenced after the Nuclear Power Plant was put in operation.

Figure 7

### Number of total Cancer Deaths per 100.000 Inhabitants

Locality of NPP Mochovce + 20 km

time periods of fuzzy c-cluster analysis



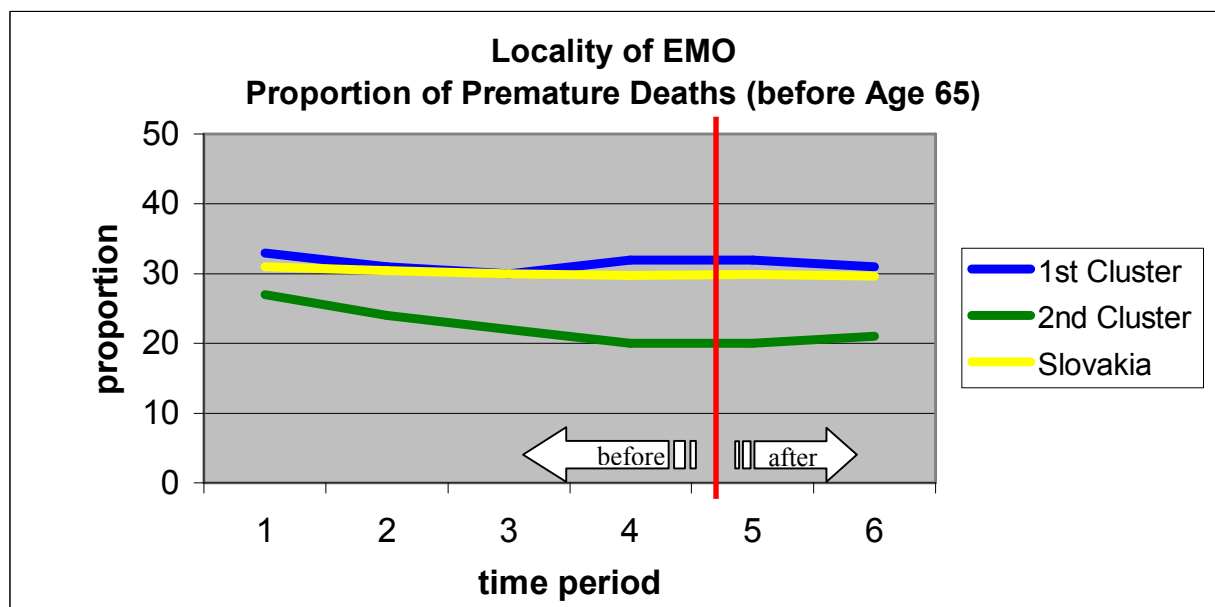
*Selected Indicator of Premature Mortality: Proportion of Premature Deaths*

Every death before the age of 65 is considered premature according to the World Health Organization (Geneva) instructions, regardless of sex.

Premature mortality in a region always signals a wider problem. Unfavourable economic situation is the most frequent factor involved. Another serious factor is environmental influence, as we can see, for example, in the vicinity of mines or huge industrial operations with high amount of exhalates.

Evaluation of the situation and tendency in the villages by means of fuzzy set of points and autocorrelation.

Figure 8



Premature deaths represent a stable proportion of all deaths in Slovakia.

It has long been observed that approximately 30 % of the nation's population die prematurely.

Our analyses suggest that there are two clusters of villages in the EMO region, both with a different situation and trend.

Both groups were close to each other around 1993 when our investigation was started. A majority of the population then began to move to the territory with the lower rate of premature deaths. We recorded a decrease by about one third in comparison with the initial situation. The second cluster of villages stagnates at the original value of 30 % with minimal decline.

Most of the villages are in the more favourable cluster with marked decrease of the number of premature deaths (dark- green) or are "rather with this condition and development" (light-green). There are larger towns mainly in the cluster with stagnation (Levice, Vráble, Zlaté Moravce).

All the values in all the villages are statistically provably accidental, which means that they are not influenced by environmental activity either in the village or in the wider area around it.

We assume, that the decline of the proportion of premature deaths is caused by markedly improved economic situation of the population thanks, in large measure, to the existence of the Mochovce nuclear power plant, which has provided many jobs and increased employment in the region.

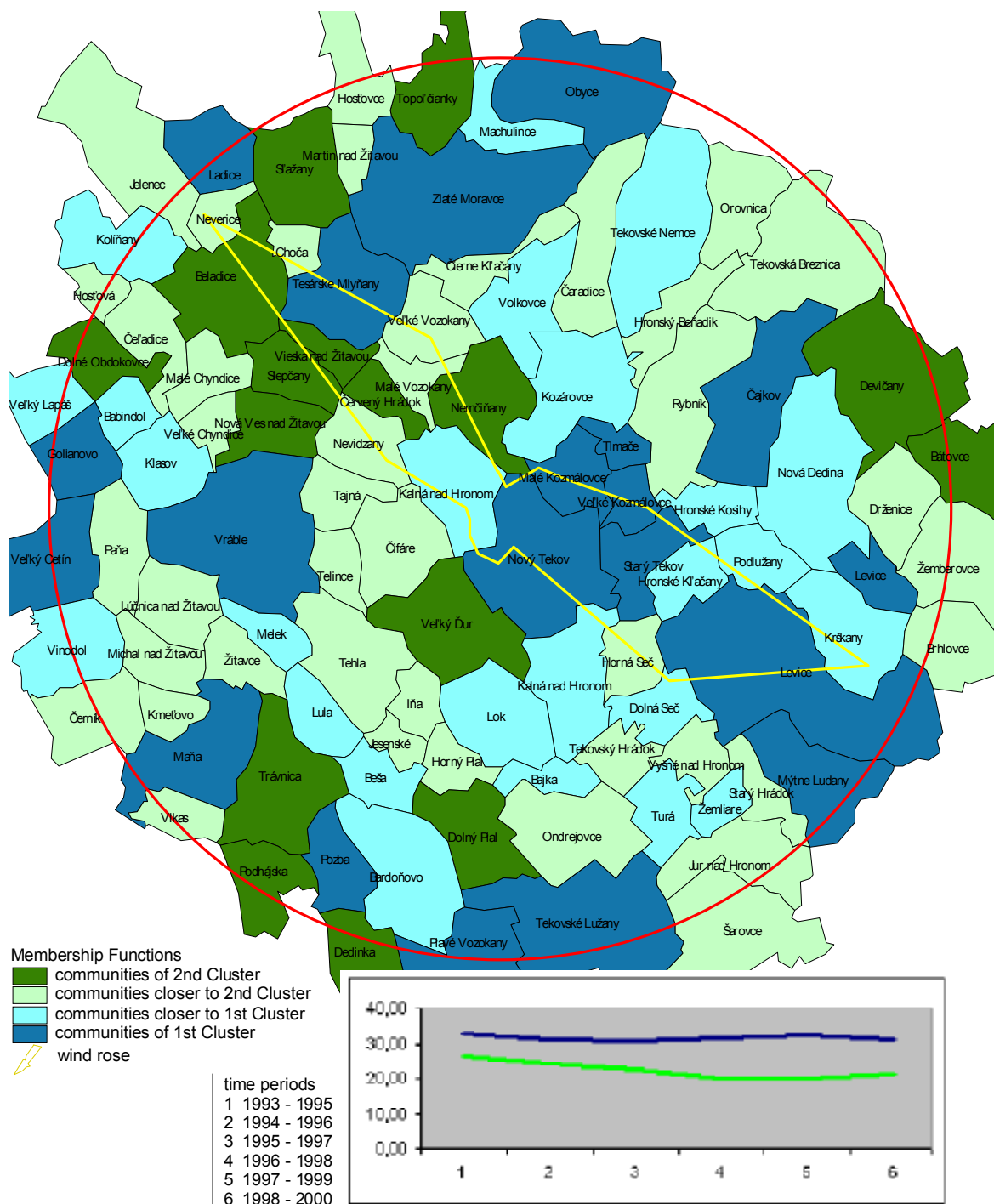


Figure 9

### Proportion of Premature Deaths (before Age 65 Years)

Locality of NPP Mochovce + 20 km

time periods of fuzzy c-cluster analysis



## *Conclusions from the evaluation of the health condition of the populations in the territory of Nuclear Power Plants*

Conclusions drawn in keeping with the individual points of our objective are as follows:

We considered the possibility of calculating all the health indicators recommended by the WHO in connection with environmental health, including detection of influence and human risks associated with industrial operation. We had chosen those indicators that are possible to calculate in Slovakia while observing the rules for determining such indicators (a total of 43 indicators).

- \* Chosen and analysed in detail were 15 indicators with at least partial relevance in all considerations of the influence of the operation of nuclear power plants

Reviewed were all available epidemiological techniques and, after careful considerations and consultations with experts at a number of institutes specialised on mathematical and medical research, we included the most up-to-date methods of polydimensional supernumerary applying methods of fuzzy set of points and space supernumerary and neuron network after thorough consultation. Our aim was to determine dependences and causes with the highest possible level of accuracy.

In the case of 15 health indicators in the territory of the "old" nuclear power plant Bohunice we concluded, that after 20 years of operation no places or localities can be found that would be at risk or where adverse impact or damage has been observed or confirmed, nor are there any territories that could be described as being at risk, either in the direction of prevailing winds or along the rivers carrying away waste water from the power plant. All the parameters are at the level found in the wider area, or even better. The entire area around the nuclear power plant seems, from the demographic point of view, to be "a town", even though it is composed of a large number of small villages. We assume, that the Nuclear Power Plant has a significant influence on the health condition of the local population, but in a positive sense. A large number of people from many local villages are employed there. These people have to lead a healthy life precisely because they work in units that have strict rules and operational discipline. Their lifestyle is similar to that of the people living in towns. They have a high income, combined with the

advantage of living in rural areas with cleaner air, lower air pollution and lower levels of exhaust fumes from motor vehicles .

In the case of the 15 health indicators in the territory of the "new" nuclear power plant Mohave we confirmed a mathematically significant influence on the whole area as well. Again in a positive sense only. There are apparent clusters of villages with low average and premature female death rates. This locality was always further away from industrial and cultural centres than the nuclear power plant Bohunice. In the past there was always scarcity of job opportunities in this region and, consequently, also greater poverty and greater morbidity. Already the construction of the power plant created now job opportunities. Even more jobs have become available with the putting of the power plant into operation and the situation keeps improving. We find this effect to be very similar to that at Bohunice: indirectly positive even if a little bit unexpected.

We can safely conclude, that objective and comprehensive evaluation of the health of the population of Slovakia is possible. Enough solid and reliable proofs are available to justify the conclusion that, regardless of the length of the power plants' operation, no unfavourable impacts on human health on their territory have been detected even by the most sophisticated research carried out by a large, multidisciplinary team of researchers from various fields of science.

# PROLONGED STRESS INDUCES ADATATION OF DROSOPHILA POPULATION TO IONIZING RADIATION

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## **Introduction**

Study on response of natural populations of living organisms to an increase in radiation background is a very important and complicated problem.

Earlier we studied genetic processes in natural populations of insects (drosophila, potatoes beetle and so on) living in areas radiocontaminated due to the Chernobyl disaster. We revealed some genetic and morphogenetic changes in these populations [1].

Since 14-15 years passed after this catastrophe, natural insect populations had to change their genetic structure because of chronic effect of radiation. It's well known that prolonged action of mutagenic factors results in adaptation of populations of living organisms to harmful agents [2-5]. Such adaptation is associated with death of more sensitive individuals and with selection and breeding of more resistant ones. It was important to investigate if natural insect populations from radiocontaminated regions became more adapted to irradiation than population from "clean" area.

## **Material and methods**

We studied natural populations of *Drosophila melanogaster* from radiocontaminated area (Vetka district of Gomel region with 24 Ci/km<sup>2</sup> of <sup>137</sup>Cs and 0,5 Cu/km<sup>2</sup> of <sup>90</sup>Sr) and from Berezynski Natural Reserve as a control area. Population samples were caught in 2000-2001 years.

Males from these two populations were exposed to 40 Gy  $\gamma$ -rays on Laboratory Cs-machine. The dose rate was 29 Gy/min. We analyzed mutations of two kinds — dominant lethal and recessive lethal mutations after additional 40 Gy irradiation in populations from different areas. In order to estimate dominant lethal mutation frequency and viability of flies, we mated

irradiated males (55 flies from each area) individually with virgin females and replaced them to bottles with fresh food daily for 3 days. We estimated the numbers of laid eggs, undeveloped eggs, alive larvae and the number of imago. Dominant lethal mutations (DLM) were estimated as a proportion of the number of undeveloped eggs to that of laid eggs.

$$\text{DLM (\%)} = \text{undeveloped egg number} / \text{laid egg number} \times 100\%.$$

Viability was estimated as a proportion of the imago number to that of laid eggs.

$$\text{Viability (\%)} = \text{imago number} / \text{laid egg number} \times 100\%.$$

Frequency of sex-linked recessive lethal mutations (RLM) was estimated by the standard method of Muller. According to this method wild-type males were mated to several virgin females of a special test line Muller-5 with mark genes “yellow” and “white apricot”. If recessive lethal mutation in X-chromosome arises, wild-type males are absent in the second generation of such mates.

Statistic processing of the experimental data was confirmed by the methods of Student and of Fisher.

## Results and Discussion

The data, obtained after irradiation of samples from natural drosophila populations, were presented in the table. DLM frequency in the control population (Berezynski Reserve) is equal to  $63,1 \pm 0,9\%$  and such parameter in Vetka district is equal to  $42,8 \pm 0,9\%$ .

**Table .** Response of drosophila populations from control and radiocontaminated areas to 40 Gy exposure (viability, DLM and RLM)

Number	Berezynski Reserve	Vetka district
Laid eggs	2831	3192
Undeveloped eggs	1786	1365
DLM (%)	$63,1 \pm 0,9$	$42,8 \pm 0,9^*$
Laid eggs	2831	3192
Imago	438	1349
Viability (%)	$15,5 \pm 0,7$	$42,3 \pm 0,9^*$
Analyzed chromosomes	831	1429
Mutations	105	95
RLM (%)	$12,6 \pm 1,1$	$6,6 \pm 0,7^*$

\* —  $p < 0,01$

Irradiation death of flies from various populations at late ontogenetic stages differs even more sharply. The imago number of population sample from control area was twice less than the laid eggs number (438 in comparison with 2831 — viability is 15,5%), but 1349 imago flies arised from 3192 laid eggs as a result of irradiation of population sample from district with high radiation background (viability makes up 42,3%).

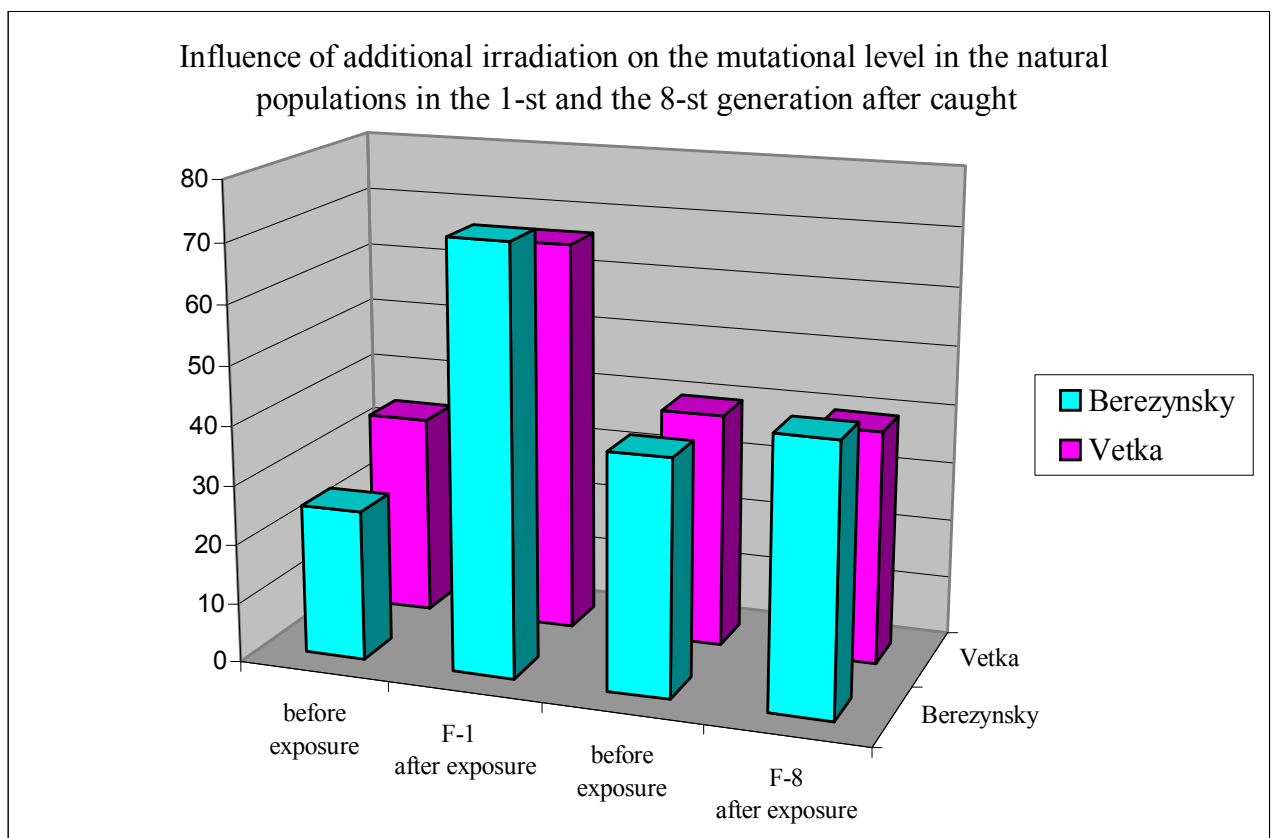
It means, that flies from radiocontaminated area are more resistant to radiation than insects from a “clean” region.

We received analogical data with help of RLM method (table). So, RLM level, induced by irradiation in the Berezynski population, was  $12,6 \pm 1,1\%$ . At the same time frequency of RLM, induced in Vetka population was  $6,6 \pm 0,7\%$ .

Thus, flies from radiocontaminated area were shown to be much more adapted to irradiation than insects from the control region.

These facts may be explained by an adaptation of exposed populations to radiation.

Then the population samples were kept under laboratory condition without irradiation. for 8 generations . It should be noted that the mutation level in both populations increased at keeping under such conditions. Acute 30 Gy irradiation was used after 8 generations. Adaptation of Vetka population to irradiation remained.. Besides the control population became also more resistant to ionizing radiation as well as Vetka population (fig.).



It means that keeping of natural drosophila populations under laboratory conditions is a strong stress (limited space, overpopulation, other than in nature temperature and light conditions), which increases mutation process and induces unspecific adaptation.

These facts should be taken into account in studying dynamics of the mutation level during radionuclide removal in animals caught in radiocontaminated regions and placed in vivarium conditions.

### **Conclusions**

1. Natural insect populations from radiocontaminated areas are more resistant to additional irradiation than control populations.
2. Keeping of natural populations under laboratory or vivarium conditions is a strong stress (limited space, overpopulation, other than in nature temperature and light conditions), which increases mutation process and induces unspecific adaptation.

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# NUMERICAL MODELLING OF LOCAL DEPOSITION PATTERNS, ACTIVITY DISTRIBUTIONS AND CELLULAR HIT PROBABILITIES OF INHALED RADON PROGENIES IN HUMAN AIRWAYS

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## **Introduction**

The general objective of our research is modelling the biophysical processes of the effects of inhaled radon progenies. This effort is related to the rejection or support of the LNT (linear no threshold) dose-effect hypothesis, which seems to be one of the most challenging tasks of current radiation protection. Our approximation and results may also serve as a useful tool for lung cancer models. In this study, deposition patterns, activity distributions and alpha-hit probabilities of inhaled radon progenies in the large airways of the human tracheobronchial tree are computed.

## **Methods**

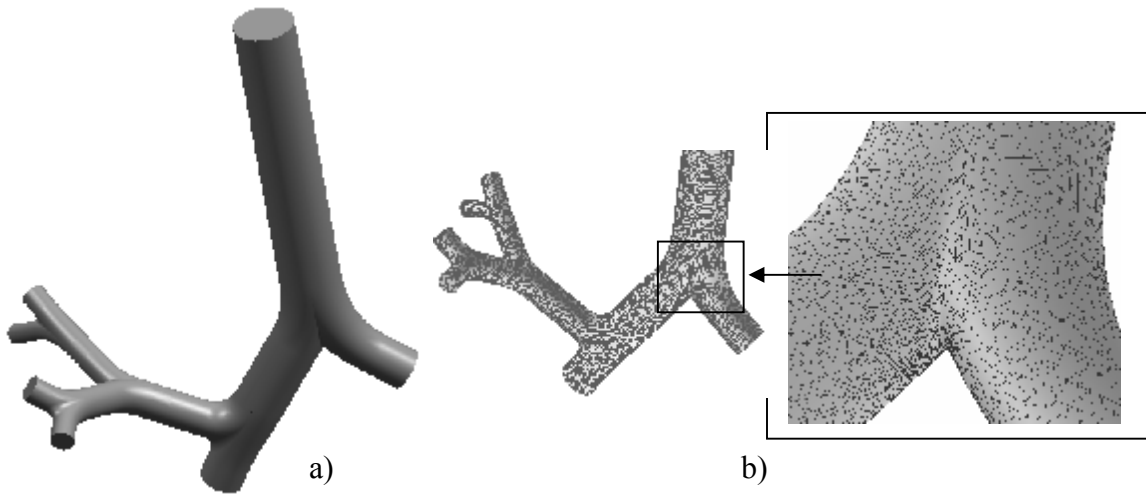
The mathematical simulation performed in this work involves a numerical model of particle deposition and a cellular hit probability-computing program.

A particle deposition model includes the morphology and geometry of the lung, gas flow structure in the inspiration-expiration cycle, and pattern of particle deposition as a result of particle-fluid interaction coupled with the geometry.

In this study, the three-dimensional physiologically realistic geometry of the generations 1-5<sup>th</sup> of the human airways (including trachea) has been generated by the Unigraphics CAD (computer aided design) software (figure 1a). The space discretization required by the applied numerical method was performed applying an unstructured, boundary and deposition density adapted tetrahedral grid (figure 1b).

For this purpose the Gambit meshing code has been used.

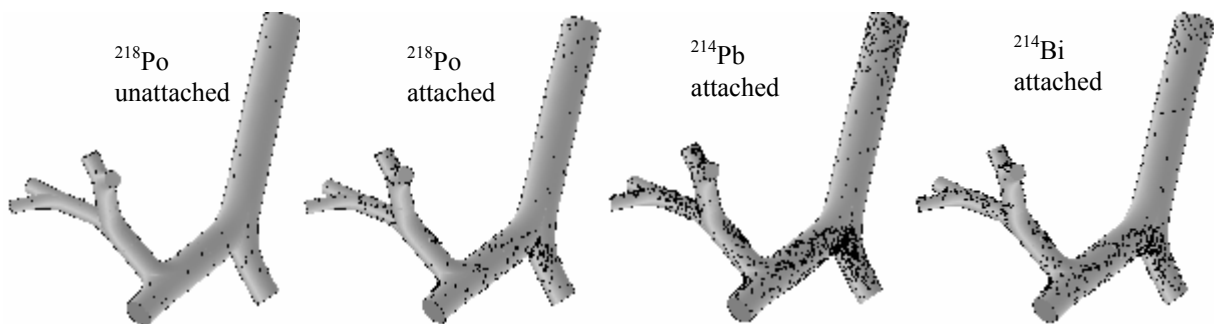




**Figure 1.** a) 3D human respiratory tract model (generations 1-5<sup>th</sup>)  
 b) the applied mathematical grid

The user enhanced FLUENT CFD (computational fluid dynamics) code has been implemented to simulate the airflow fields for different flow rates and radioisotopes during inspiration and expiration. Trajectories of injected spherical particles have been computed by consideration of basic deposition mechanisms, that is the inertial impaction, sedimentation and Brownian diffusion. Both attached ( $^{218}\text{Po}$ ,  $^{214}\text{Bi}$  and  $^{214}\text{Pb}$ ) and unattached ( $^{218}\text{Po}$ ) radionuclids have been tracked taking into account their characteristic sizes and concentrations (figure 2).

Regarding the particle deposition two assumptions was made: wall-impacting particles was considered deposited and interception (which is an important deposition mechanism in case of fibrous particles) was neglected.

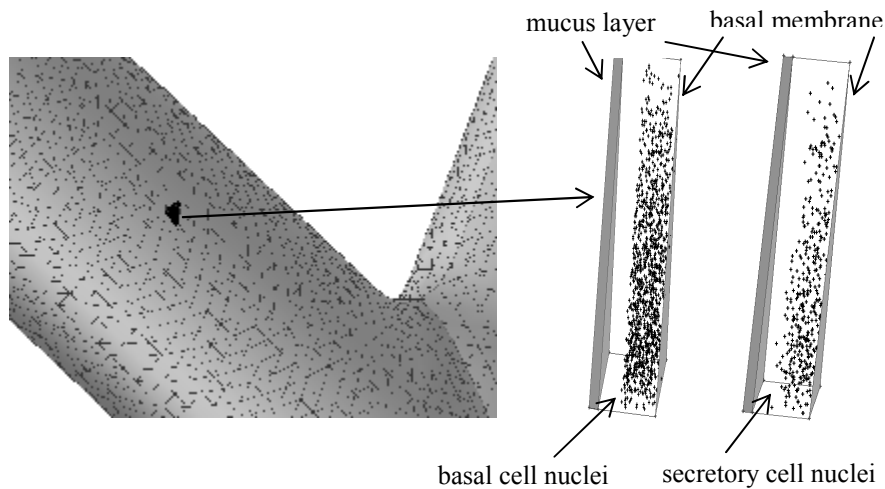


**Figure 2.** Deposition pattern of attached and unattached radon progenies.

Activity distributions of deposited radionuclides were quantified on the basis of the resulted deposition patterns in circumstances characteristic in uranium mines and homes, taking into account that  $^{218}\text{Po}$  (which represents  $\sim 13\%$  of the attached and  $\sim 100\%$  of the attached fraction) is (potentially) a double alpha particle emitter.

Hit probabilities of alpha particles emitted by deposited radon progenies in the cell nuclei of the tracheobronchial epithelium were computed by an own code using realistic epithelial cell morphology model. The code uses as input data the deposition pattern and the lung geometry, generates the alpha sensitive epithelial cells and alpha particle tracks then computes the single and multiple cell nuclei hits.

The epithelial cell nuclei (basal, secretory, goblet, ciliated and preciliated) are generated on the basis of the depth distribution functions taken from the literature and built in the code (figure 3).



**Figure 3.** Distribution of basal and secretory cell nuclei in one computational grid cell.

The decay directions are randomly selected using a Monte-Carlo technique, leading to the near wall and far wall components. The two different alpha tracks have different lengths because of the different energy loss in air and tissue (figure 4). Delivered dose computations are also planned by the inclusion of a cord length computing routine, the dose being:

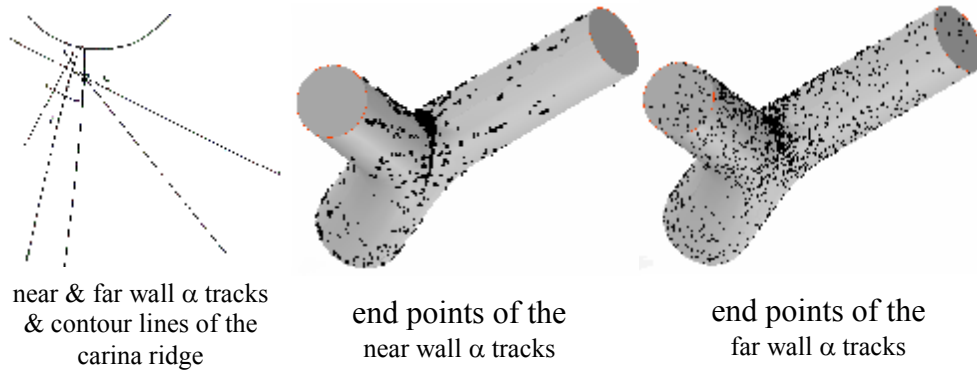
$$D = \frac{E}{m} \quad (1)$$

, where  $D$  is the transmitted dose,  $E$  the transmitted energy and  $m$  the mass.

$E$  can be computed from the

$$E = l \cdot LET \quad (2)$$

equation, where  $l$  is the cord length and  $LET$  the linear energy transfer.



**Figure 4.** Some near and far wall  $\alpha$  tracks at a carinal ridge of a bifurcation (left panel), end points of about 2000 near wall (middle panel), and 2000 far wall (right panel) alpha tracks in the epithelium of one airway bifurcation emitted by deposited attached radon progenies.

### Results and conclusions

The airflow fields and related particle deposition patterns strongly depend on the shape of airway geometry and breathing pattern. Computed deposition patterns of attached and unattached radon progenies are strongly inhomogeneous creating hot spots at the carinal regions and downstream of the inner sides of the daughter airways. The results suggest that in the vicinity of the carinal regions the multiple hit probabilities are quite high even at low average doses and increase exponentially in the low-dose range. Thus, even the so-called low doses may present high doses for large clusters of cells. The cell transformation probabilities are much higher in these regions and this phenomenon cannot be modeled with average burdens.

### Acknowledgements:

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# BIODOSIMETRY OR BIOINDICATION?

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## **Introduction**

Humans have different radiosensitivity, which is quantitative character and its distribution in a population corresponds to binomial law.. So radiosensitivity of a person can differ from the medium value by 2-3 sigma [1]. It means that reaction of people on the same radiation dose is different. Radiosensitivity are caused by interaction of some pairs of polymeric genes determining a number of physiological and biochemical organism features.. Besides, radiosensitivity as well as all other quantitative characters depends to a great degree upon environmental factors. That's why many factors which are not under control can change significantly biological effects of radiation and by this can be responsible for serious mistakes of biodosimetry.

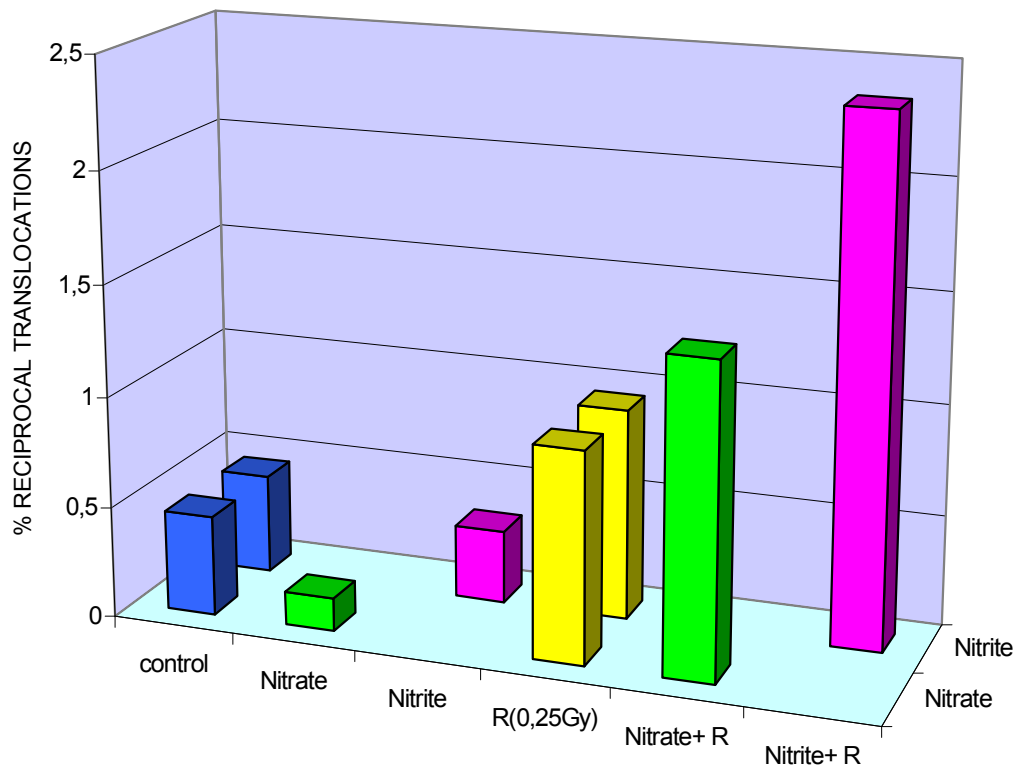
## **Fertilizers and herbicides influence**

Some of substances are present in our food — residual amounts of fertilizers or herbicides can be mutagenic or influence mutagenic action of radiation. In the last case synergetic or antagonistic effects can be observed.

We studied cytogenetic effects of such fertilizers as sodium nitrite and nitrate and herbicide zenkor in mice and human cells. Reciprocal translocations in germ cells of mice were cytologically analysed in metaphase of spermatocytes by the method, which is a modified Ivens' method. Human cells were cultured according to a standart method, the cytological test included dicentric-ring-fragment analysis.

Sodium nitrite and nitrate in concentrations of 10 and 100 mg/l respectively were added to drinking water for mice within 2,5; 5,0 and 7,5 months. It was found that these substances didn't possess mutagenic activity under such conditions. Acute or prolonged irradiation of treated mice with gamma-rays or neutrons 0,25; 0,5 and 5,0 Gy showed that  $\text{NaNO}_2$  and  $\text{NaNO}_3$  sensibilized significantly (2-4 times) cytogenetic action of ionizing radiation (fig.1). We have found the same effect in drosophila earlier.

**Fig.1 INFLUENCE OF SODIUM NITRITE AND SODIUM NITRATE ON MUTAGENIC ACTION OF RADIATION IN MICE GERM CELLS**



Cytogenetic effects of zenkor were found to be completely different. This herbicide being injected intraperitoneally or perorally to mice increases the translocation rate in germ cells and being added to culture media for human cells shows a mutagenic effect too. Irradiation of treated mice and human cells results in decreasing aberration levels on the contrary with nitrite and nitrate. Chronic influence of zenkor and gamma-rays leads to more strict “antagonistic” effect than acute one [2]. Such effect can be explained by increased death of cells treated with both mutagenic factors — zenkor and irradiation. This explanation had been proved by our experiments with drosophila populations, in which combined action of these factors resulted in decreasing flies number. Thus different agricultural chemicals have different cytogenetic effects used in combination with irradiation.

### **Radioprotectors and radioadaptive response influence**

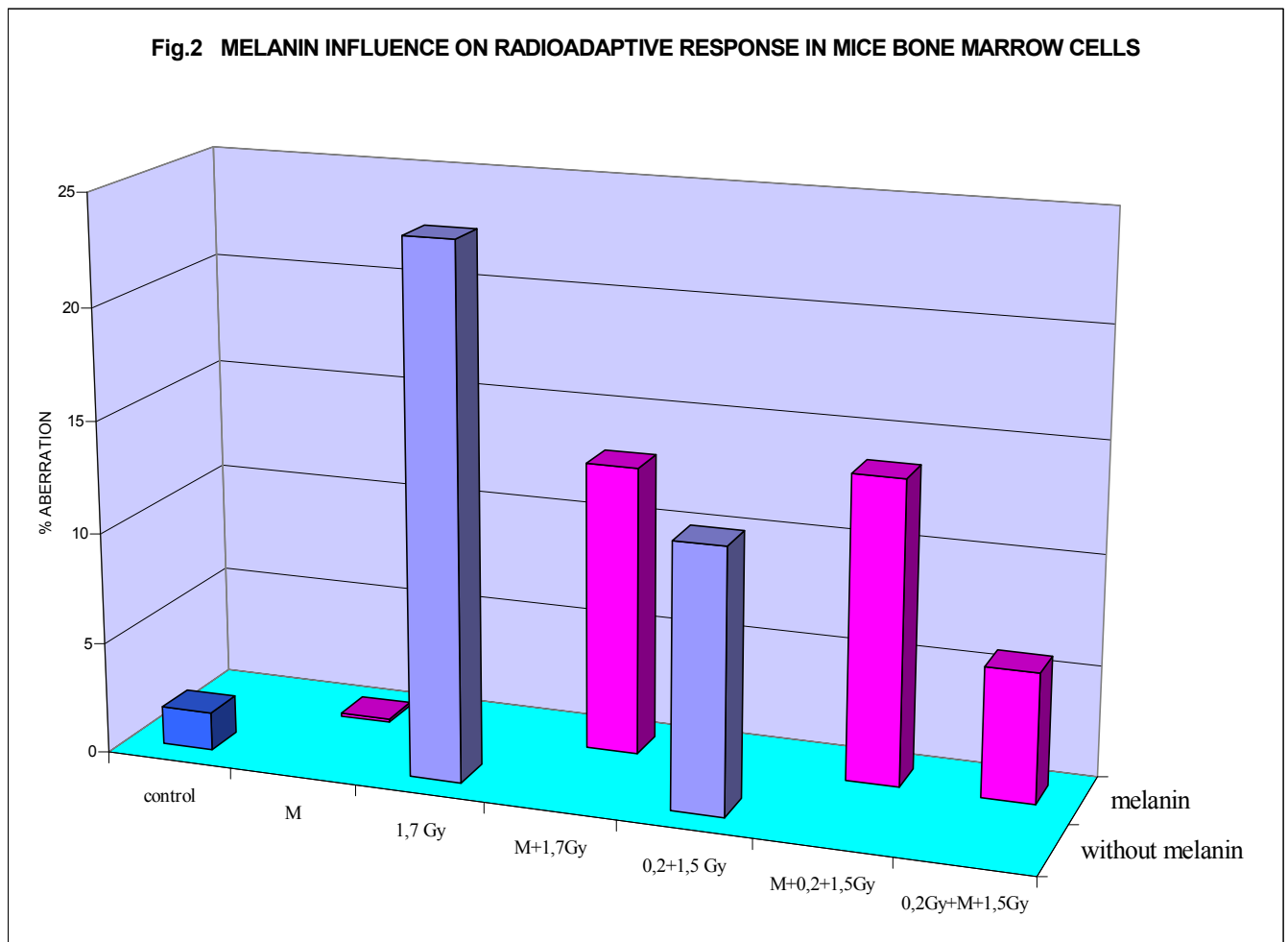
In the same time many food stuffs contain radioprotectors or antimutagens. So, tea, coffee, cocoa, chocolate, mushrooms and others have melanin, which is very effective radioprotector not only against acute irradiation, but even against chronic one [3].

Thus, some chemicals can influence results of biodosimetry strongly. Some drugs, stress, virus diseases and so on can change biological effects of radiation too.

Radioadaptive response is one of the most significant factors which can be responsible for uncorrect radiation dose evaluation.

This phenomenon decreases effects of ionizing radiation approximately twice. Adaptive reaction can be induced by low radiation dose as well as by weak chemical mutagens and many other factors [4].

We demonstrate adaptive response in mice germ ad bone marrow cells – rate of chromosomal aberrations after 0,2 + 1,5 Gy was about twice less than after 1,7 Gy irradiation. Melanin influence on adaptive response has been studied. Melanin injection 2 hours before the first conditioning dose 0,2 Gy resulted in the same mutation level as before 1,7 Gy – adaptive response was not found. If melanin was applied between the first and the second doses both adaptive phenomenon and melanin protection led to 4-fold decrease in aberration rate (fig.2). So, effective radioprotectors are able to prevent adaptive response by suppressing conditioning dose effect or to decrease some times genetic effect of radiation.



All these uncontrolled factors may be responsible for significant mistakes of evaluating single radiation dose and the estimation of chronic radiation dose has much more problems. It is necessary to understand that biological methods show only a level of biological consequences of irradiation, but they do not allow to estimate exposed or absorbed radiation doses. That's why biological methods can be used for bioindication but not for biodosimetry.

### **Conclusions**

1. Humans have different radiosensitivity, It means that reaction of people on the same radiation dose is different.

2. Some food stuffs contain residual amounts of fertilizers or herbicides which can be mutagenic or influence mutagenic action of irradiation. Another ones contain radioprotectors. For instance, tea, cocoa, coffee, mushrooms contain melanin, which decreases effectively biological damage, induced not only by acute irradiation, but even by chronic one

3. Adaptive response decreases effects of ionizing radiation approximately twice. This reaction can be induced by low radiation dose as well as by weak chemical mutagens and many other factors

4. Many other factors (drug use, stress, virus diseases etc.) can influence biological action of radiation.

5. All these uncontrolled factors may be responsible for significant mistakes of evaluating radiation doses. That is why biological methods can be used for bioindication but not for biodosimetry.

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# **BIOMONITORING OF GENOTOXIC RISK IN RADAR FACILITY WORKERS: COMPARISON OF THE COMET ASSAY WITH MICRONUCLEUS ASSAY AND CHROMATID BREAKAGE ASSAY**

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## **Abstract**

Genotoxic risks of occupational exposure in a radar facility were evaluated by using alkaline comet assay, micronucleus assay and chromatid breakage assay on peripheral blood leukocytes in exposed subjects and corresponding controls. Results show that occupational exposure to microwave radiation correlates with an increase of genome damage in somatic cells. The levels of DNA damage in exposed subjects determined by using alkaline comet assay were increased compared to control and showed interindividual variations. Incidence of micronuclei was also significantly increased compared to baseline control values. After short exposure of cultured lymphocytes to bleomycin, cells of occupationally exposed subjects responded with high numbers of chromatid breaks. Although the level of chromosome damage generated by bleomycin varied greatly between individuals, in exposed subjects a significantly elevated number of chromatid breaks was observed. Our results support data reported in literature indicating that microwave radiation represents a potential DNA-damaging hazard. Alkaline comet assay is confirmed as a sensitive and highly reproducible technique for detection of primary DNA damage inflicted in somatic cells. Micronucleus assay was confirmed as reliable biomarkers of effect and chromatid breakage assay as sensitive biomarker of individual cancer susceptibility. The results obtained also confirm the necessity to improve measures and to perform accurate health surveillance of individuals occupationally exposed to microwave radiation.



## **Introduction**

The people of industrialised societies are continuously exposed to increasing levels of electromagnetic fields (EMF) emitted by various electrical installations and telecommunication systems [1]. In recent years there has been growing interest in the health effects of the electromagnetic radiation's designated extremely low frequency (ELF) and radiofrequency radiation (RFR). There is evidence that microwaves cause different biological effects depending upon field strength, frequencies, wave forms, modulation and duration of exposure. These effects were mainly attributed to microwave heating but recent reports have shown or suggested that there are nonthermal microwave effects in terms of energy required to produce various types of molecular transformations and alterations [2]. It is known that exposure to microwave radiation has different biological effects on eye, the nervous system and its function, circulatory and the reproductive system [3,4]. Reports on cytogenetic risks from microwave exposure on the induction of chromosome damage are sometimes contradictory, mainly because of different experimental conditions of *in vitro* and *in vivo* studies [5]. However, in occupationally exposed subjects increased levels of DNA damage as expressed by means of cytogenetic endpoints were observed [3, 6-8]. Increased incidence of micronuclei is also reported after *in vitro* exposure to microwave radiation on human lymphocytes [9]. It has been suggested that exposure to radiofrequency radiation may have genetic effects which predispose to the development of cancer, particularly lymphoma and leukaemia, and also birth defects such as Down's syndrome [10-11].

Biomonitoring of human populations exposed to potential mutagens / carcinogens is an early warning system for genetic diseases or cancer. It also allows detection of different risk factors at a time when control measures still could be implemented [12]. For the detection of early biological effects of DNA-damaging agents in environmental and occupational settings a wide range of methods is presently used. Among them, there are well-established cytogenetic biomarkers such as structural chromosomal aberrations (CA), micronuclei (MN) and sister chromatid exchanges (SCE) [13,14,15]. During the last decade, the comet assay or single cell gel electrophoresis (SCGE) was introduced as a useful technique for human biomonitoring studies [16,17]. Contrary to cytogenetic biomonitoring techniques that are limited to circulating lymphocytes and proliferating cell populations, the comet assay can be applied both to proliferating and non-proliferating cells. Moreover, the comet assay is a rapid, simple and

sensitive technique for measuring DNA breakage with a small number of cells and detects intercellular differences in DNA damage [18,19].

Long term occupational exposure to physical and chemical mutagens often is related with an elevated risk for DNA damage. Most exposures in an occupational setting are chronic or repeated and the induced DNA damage might be expected to be elevated above baseline levels. The individuals with even slight deficiency in any step of DNA repair can accumulate mutations, and consequently their cancer risk is elevated. The simple and inexpensive test, as chromatid breakage assay, which allows selection of persons with a defect in DNA repair, could be a useful additional marker in biomonitoring studies [20]. Susceptibility to bleomycin-induced chromatid breaks in cultured peripheral blood lymphocytes may reflect the way a person deals with carcinogenic challenges [21].

The objective of the present study was to evaluate the levels of primary DNA damage, chromosomal damage and mitotic spindle disturbances as well as the mutagen sensitivity in peripheral blood leukocytes of radar-facility workers daily exposed to microwave radiation compared to unexposed control subjects. As sensitive biomarkers the alkaline comet assay, micronucleus assay and chromatid breakage assay (bleomycin sensitivity test) were selected.

### **Subjects and methods**

The study was carried out on 10 clinically healthy male workers, employed on radar equipment and antenna system service. They worked within a microwave field of  $10 \text{ (W/cm}^2 - 20 \text{ mW/cm}^2)$  with frequency range of 1250-1350 MHz. Their average age was 50.1 yr (range 45 - 56 yrs) and mean employment duration 24.3 yrs (range 7-29 yrs). Two subjects were smokers, three of them were ex-smokers and five non-smokers. The control group consisted of 10 subjects of similar age. Three of them were smokers, two ex-smokers and five non-smokers. All subjects were questioned concerning age, their presented medical history, family history on cancer, lifestyle, occupational and extraoccupational exposure to potential mutagenic hazards. Samples of peripheral blood were collected by venipuncture into heparinized tubes in the morning. The heparinized blood was then coded and stored in the refrigerator ( $4^\circ\text{C}$ ). The alkaline comet assay and initiation of in vitro blood cultivation for the cytogenetic endpoints and chromatid breakage assay were performed within a maximum of 2 h period after blood collection.

**The alkaline comet assay** was carried out under alkaline conditions, basically as described by Singh et al. [16]. Fully frosted slides were prepared by covering with 1 % and 0.6 % normal melting point (NMP) agarose (Sigma). After solidification, a layer containing the whole blood sample mixed with 0.5 % low melting point (LMP) agarose (Sigma) was put on the slide and afterwards covered with 0.5 % LMP agarose. Slides were immersed for 1 h in ice-cold freshly prepared lysis solution (2.5 M NaCl, 100 mM Na<sub>2</sub>EDTA, 10 mM Tris-HCl, 1 % Na-sarcosinate (Sigma), pH 10) with 1 % Triton X-100 (Sigma) and 10 % dimethyl sulfoxide (Kemika). The slides were then placed on a horizontal gel-electrophoresis tank, facing the anode. The unit was filled with fresh electrophoretic buffer (300 mM NaOH, 1 mM Na<sub>2</sub>EDTA, pH 13.0). Denaturation (20 min) and electrophoresis (20 min at 25 V or 300 mA) were performed at 4°C under dim light. After electrophoresis the slides were rinsed gently three times with a neutralisation buffer (0.4 M Tris-HCl, pH 7.5) to remove excess alkali and detergents. Each slide was stained with ethidium bromide (20 µg/ml) and covered with a coverslip. A total of 100 randomly captured comets per sample were examined at 400 x magnification using an epifluorescent microscope (Zeiss) connected through a black and white camera to an image analysis system (Comet Assay II, Perceptive Instruments Ltd., U.K.). To quantify the DNA damage, tail length (TL), and tail moment (TM) were evaluated. Tail length (length of DNA migration) is related directly to the DNA fragment size and presented in micrometers. It was calculated from the centre of the cell. Tail moment was calculated as the product of the tail length and the fraction of DNA in the comet tail. All measurements were summarized as the mean ± SE. Measured comet parameters were evaluated by using analysis of variance and non-parametric Mann-Whitney U test. The level of statistical significance was set at  $p < 0.05$ .

**Lymphocyte culture** - For each blood sample double cultures were set up. Lymphocytes were grown for 72 h at 37 °C in F-10 medium (Sigma, USA), supplemented with 10 % foetal calf serum (Biological Industries, Israel), phytohaemagglutinin (Murex Biotech Ltd., England) and antibiotics penicillin (Crystacillin<sup>®</sup>, Pliva, Croatia) and streptomycin (Streptomycin sulfat, Krka, Slovenia).

**Micronucleus assay.** The micronucleus assay was performed as described by Fenech and Morley [15] with some modifications. Cytochalasin-B (Sigma, USA) at final concentration of 6 (µg /ml) was added to each sample at 44 h and the cells were harvested after a further incubation of 28 h. After a brief treatment with physiological saline, cells were fixed with 3:1 mixture of methanol

and acetic acid. Permanent preparations were obtained by pipetting a few drops of the cell suspension onto clean slides. Slides were air-dried overnight at room temperature and stained with 5 % Giemsa (Sigma, USA) solution at pH 6.8 for 10 minutes. After staining, slides were washed and air-dried. Incidence of MN was evaluated by scoring of 500 binucleated cells. Total number and distribution of MN was determined. The statistical significance of differences between frequencies of MN in exposed and control subjects was tested by using the  $\chi^2$  test. The level of significance was set at  $p < 0.05$ .

***The chromatid breakage assay*** - Duplicate cultures were set up for each subject. The cultures were set up by adding 0.5 mL of whole heparinized blood to RPMI 1640 medium (Euroclone), supplemented with 10 % fetal calf serum and antibiotics. After the lymphocyte cultures were incubated for 3 days at 37°C, they were incubated for 5 hours with bleomycin sulfate (30 µg/ml; Nippon Kayaku Co). To arrest the cells at metaphase, 0.04 µg/ml colchicine (Sigma Chemical Co) was added to the cultures 1 hour before harvesting. The cells were harvested for chromosome analysis by standard procedures, and slides were made by the conventional air-drying technique. Metaphase spreads were stained with 5 % Giemsa solution (Merck). 200 metaphase spreads per subject were scored blindly for the presence of chromatid breaks. Chromatid breaks were recorded by following previously established criteria [21]. The mean number of breaks per cell was used as a measure for the individual susceptibility. Differences between groups with respect to mutagen sensitivity were assessed by use of Mann-Whitney U test.

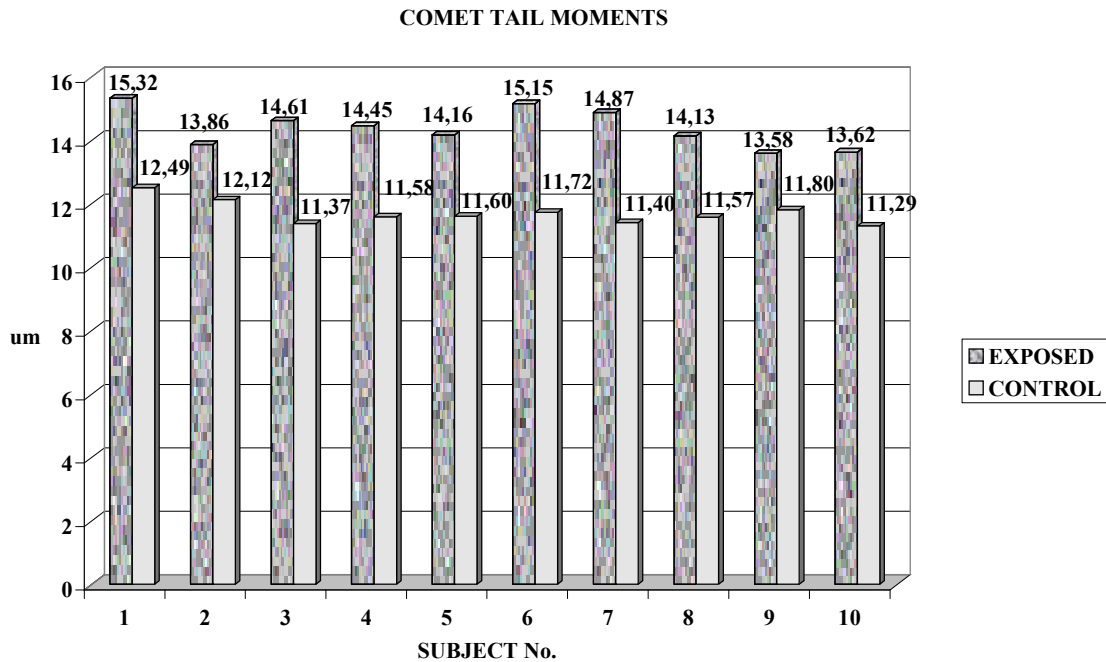
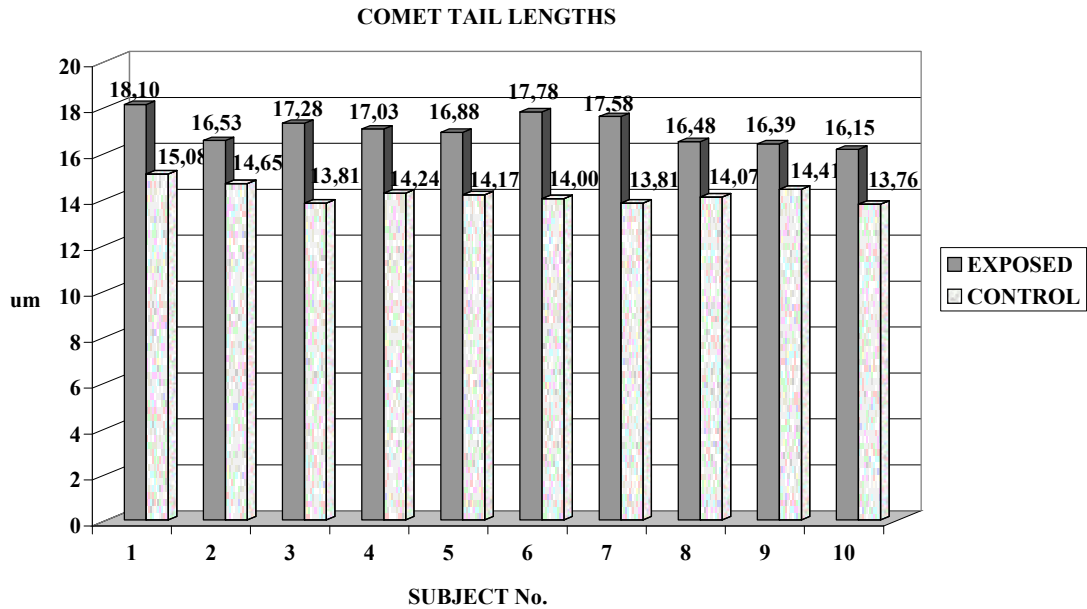
## **Results**

### ***The alkaline comet assay***

Results of the alkaline comet assay are shown in Figure 1. Between the exposed and control groups statistically significant differences in both comet parameters evaluated were observed. In blood samples collected from the exposed subjects significant interindividual differences in DNA migration pattern regarding both comet parameters were recorded ( $p < 0.05$ , the analysis of variance), with a shift to the upper level of damage compared to control. For the exposed group the mean value of comet tail length was  $17.06 \pm 0.10$  µm (range 12.18 µm - 53.85 µm), while a mean value of comet tail moment was  $14.40 \pm 0.09$  (range 9.44 - 47.04). The highest DNA damage, as determined by means of comet tail length and tail moment was found to be in the subject No. 1. The observed DNA damage was found to be the lowest in the subject No. 10, however it was still significantly elevated compared to values of comet parameters measured

in control healthy blood donors (Figure 1). When compared occupationally exposed smokers (ex-smokers included) and non-smokers there were no significant differences regarding both comet parameters observed ( $p>0.05$ , Mann-Whitney  $U$  test). On the other hand, within the blood samples collected from control subjects significantly lower values of both comet parameters studied were recorded (Figure 1). Mean value of comet tail length was  $14.20 \pm 0.05 \mu\text{m}$  (range  $10.90 \mu\text{m} - 18.95 \mu\text{m}$ ), while a mean value of comet tail moment was  $11.70 \pm 0.05$  (range  $8.26 - 16.03$ ). However, unexposed smokers (ex-smokers included) had significantly higher levels of overall DNA damage than non-smokers, as expressed by means of both comet parameters measured ( $p<0.05$ , Mann-Whitney  $U$  test). In unexposed smokers mean tail length was  $14.39 \pm 0.07 \mu\text{m}$ , and mean tail moment  $11.93 \pm 0.07$ , while in non-smokers mean tail length was  $14.01 \pm 0.06 \mu\text{m}$ , and mean tail moment  $11.56 \pm 0.06$ .

Figure 1. Results of the alkaline single cell gel electrophoresis on the peripheral blood leukocytes from 10 radar-facility workers daily exposed to microwave radiation and 10 unexposed control subjects.



### Micronucleus assay

Results regarding the frequency, distribution and total number of micronuclei (MN) for the exposed and control group are reported in Table 1. In exposed group there were statistically significant increases in total number of micronuclei recorded ( $p < 0.05$  using  $\chi^2$  test) compared to control. Distribution of micronuclei per 500 binucleated cells in all exposed subjects was also disturbed compared to control (Table 1).

Table 1. Distribution and total number of micronuclei (MN) in peripheral blood lymphocytes from microwave-exposed subjects and corresponding controls.

Subject smoking / exposure	Distribution of MN				Total No. of MN / 500 cells
	0	1	2	3	
<b>EXPOSED GROUP</b>					
<b>1</b> (S; 22)	489	9	2	0	13
<b>2</b> (S; 29)	484	15	1	0	17
<b>3</b> (ES; 28)	478	19	3	0	25
<b>4</b> (ES; 26)	484	15	0	1	18
<b>5</b> (ES; 26)	482	16	2	0	20
<b>6</b> (N; 7)	490	10	0	0	10
<b>7</b> (N; 26)	472	13	4	1	24
<b>8</b> (N; 25)	471	19	0	0	19
<b>9</b> (N; 29)	489	11	0	0	11
<b>10</b> (N; 25)	479	9	2	0	11
<i>Mean ± SD</i>					<b>16,8 ± 5,4<sup>-</sup></b>
<b>CONTROL GROUP</b>					
<b>1</b> (S)	495	3	1	0	5
<b>2</b> (S)	494	6	0	0	6
<b>3</b> (S)	495	5	0	0	5
<b>4</b> (ES)	497	3	0	0	3
<b>5</b> (ES)	494	5	1	0	7
<b>6</b> (N)	496	3	1	0	5
<b>7</b> (N)	496	4	0	0	4
<b>8</b> (N)	496	4	0	0	4
<b>9</b> (N)	494	6	0	0	6
<b>10</b> (N)	497	3	0	0	3
<i>Mean ± SD</i>					<b>4,8 ± 1,3</b>

-  $p < 0.05$  compared to control ( $\chi^2$  test).

*The chromatid breakage assay*

The results of the chromatid breakage assay are summarised in Table 2. In occupationally exposed subjects chromatid breaks were found in 95-179 cells out of 200 metaphases scored (mean value of 123 damaged metaphases per subject). Most frequently there were 1 to 4 breaks in one damaged metaphase, frequently 5 to 7 breaks, and less frequently 8 to 11 breaks in one damaged metaphase. The multiple aberrations (>11 breaks in one cell) were seldom. The highest number of chromatid breaks scored per metaphase was 30, and it was recorded in two subjects (Nos. 2 and 4). The average number of breaks per cell (b/c) per person varied widely between exposed subjects, from a low value of 0.88 to as high as 3.66 (Table 2). The distribution of the b/c values recorded was shifted towards higher values compared to control. In control subjects chromatid breaks were found in 34-59 cells out of 200 metaphases scored (mean value of 44 damaged metaphases per subject). Most frequently there were 1 to 2 breaks in one damaged metaphase, frequently 3 to 4 breaks. The multiple aberrations were seldom. The highest number of chromatid breaks scored per metaphase was 14, and it was recorded in only one subject (No. 9). The average number of b/c per person varied widely between exposed subjects, from a low value of 0.39 to as high as 0.61 (Table 2). The differences in b/c values recorded between smokers and non-smokers in both study groups were not statistically significant.

Table 2. Distribution of results of the chromatid breakage assay on the peripheral blood lymphocytes from microwave-exposed subjects and corresponding controls.

Subject No.	EXPOSED GROUP			CONTROL GROUP		
	Metaphases with breaks (no.) <sup>*</sup>	Mean (± SE) b/c	Min. – Max. b/c	Metaphases with breaks (no.) <sup>*</sup>	Mean (± SE) b/c	Min. – Max. b/c
<b>1</b>	114	1.61 ± 0.17	1-16	34	0.53 ± 0.11	1-11
<b>2</b>	179	3.66 ± 0.28	1-30	46	0.54 ± 0.10	1-11
<b>3</b>	95	0.88 ± 0.11	1-14	41	0.55 ± 0.11	1-12
<b>4</b>	144	2.27 ± 0.22	1-30	38	0.39 ± 0.07	1-7
<b>5</b>	114	1.22 ± 0.12	1-8	40	0.54 ± 0.11	1-13
<b>6</b>	127	1.46 ± 0.13	1-11	50	0.61 ± 0.11	1-10
<b>7</b>	100	0.88 ± 0.09	1-9	59	0.54 ± 0.09	1-11
<b>8</b>	154	2.86 ± 0.24	1-21	46	0.53 ± 0.09	1-7
<b>9</b>	98	0.97 ± 0.12	1-16	47	0.59 ± 0.12	1-14
<b>10</b>	103	1.10 ± 0.11	1-11	38	0.40 ± 0.08	1-10
<b>Mean (± SE)</b>	<b>123</b>	<b>1.69 ± 0.06-</b>	<b>1-17</b>	<b>44</b>	<b>0.52 ± 0.03</b>	<b>1-14</b>

b/c – breaks per cell; <sup>\*</sup>200 metaphases per subject scored; - p<0.05 compared to control (Mann-Whitney *U* test)



## **Discussion**

Almost every member of our technologically modern society is exposed to some nonionizing radiation. However, the intensity of these exposures in the general population is very low compared to exposures in occupational settings, where people may work near various sources [26]. Many epidemiological and experimental studies have been carried out to investigate the possible health hazards associated with human exposure to ELF or RF electromagnetic fields. Various studies have provided evidence suggesting that EMFs with relatively low intensity are capable of interacting with many molecular, cellular and systemic processes associated with carcinogenesis, mutagenesis and teratogenesis [1]. Biologically detrimental effect of the forementioned radiation in occupationally exposed persons has not yet been sufficiently explored and the existing data are often contradictory. Protection of the radar-exposed personnel is complicated due to the fact that there are no suitable personal dosimeters similar to those carried by subjects occupationally exposed to ionizing radiation. Therefore, using of different cytogenetic biomarkers become very important. In the present study potential genotoxic effects of the long-term occupational exposure to microwave radiation were evaluated on subjects daily employed on radar equipment and antenna system service. As sensitive biomarkers of the exposure, effect and susceptibility, the alkaline comet assay, micronucleus assay and chromatid breakage assay (bleomycin sensitivity test) were selected. For human biomonitoring of environmental and occupational exposure to mutagens and / or carcinogens different chemical and biological endpoints are used. Cytogenetic endpoints such as chromosomal aberrations, micronuclei and sister chromatid exchanges in peripheral blood lymphocytes have been extensively employed as cytogenetic biomarkers to assess genotoxic risks in the occupational settings. It has been found out that increased cytogenetic damage reflects an enhanced cancer risk [14]. During the last decade, the comet assay or single cell gel electrophoresis (SCGE) was also introduced as a sensitive and reliable biomarker in human biomonitoring studies [11,15-17].

Present study reports the results of the application of alkaline comet assay on peripheral blood leukocytes in long-term radar-exposed workers. Although the mechanism by which RFR affects DNA is not known, and the results of the reported studies on genotoxic effects of RFR exposure are often conflicting [4], our results indicate potential of RFR to induce measurable DNA-damaging effects in long-term exposed personnel. Our results indicate that the alkaline comet assay can be successfully applied to the study of microwave-induced primary DNA damage and repair in peripheral blood leukocytes of exposed subjects. By using the alkaline

comet assay in exposed subjects a lot of DNA single strand breaks and alkali-labile sites were detected. Similar results reported Lai and Singh [21] on rat brain cells after acute exposure to radiofrequency electromagnetic radiation. Their results indicate that acute exposure to RFR at an average body SAR of 1-2 W/kg caused a significant increase in both single- and double- strand DNA breaks. By using other cytogenetic endpoints it was also recorded earlier that similar occupational settings are associated with increased chromosome damage [5,6]. It should be pointed out that the type of DNA damage recorded by alkaline comet assay is rather continuously and efficiently repaired. Therefore, the measured damage level is a result of equilibrium between damage infliction and repair. It is known that DNA single strand breaks are rapidly repaired, however, other lesions, such oxidized bases may persist longer and be misrepaired to somatic mutations [15]. On the other hand, apurinic/apyrimidinic sites (alkali-labile sites) may not be readily repaired in  $G_0$  cells and without DNA replication they may prove to be “silent” DNA lesions [22]. It is known that DNA damage accumulates with time and the repair capacity decreases during long-term occupational exposure. Therefore, when DNA damage is improperly repaired or is not repaired, different biological effects may result. Although majority of DNA lesions is repaired in a few hours and days following their infliction, a part of DNA damage induced persisted over time. It can be considered that this elevated level reflects an accumulation of non-repaired DNA damage. High level of persistent DNA damage can lead to cell death. Large numbers of cells dying can lead to serious organ failures while damaged or improperly repaired DNA may develop into cancers. Reports on the possible cancer-related effects of RFR are still controversial. However, it is possible that in complex cellular processes involved in carcinogenesis it could have co-carcinogenic effect [7]. Differences in an individual’s capability to deal with mutagenic assault can be measured using chromosomal aberration tests. In addition to measuring chromosomal aberrations in lymphocytes, the analysis of micronuclei in binucleate cytokinesis blocked lymphocytes has also been suggested for biological dosimetry. It is thought that any aberration or even a whole chromosome might become micronuclei. Our results also confirmed that enumeration of micronuclei provides a reliable index of chromosome loss from the main nuclei indicating clastogenic and aneugenic impacts of long-term occupational exposure to microwave radiation. Using the bleomycin-based mutagen sensitivity assay in present study radar-exposed and control subjects have been screened. The level of chromosome damage generated by bleomycin varies greatly between individuals. Our results indicate that observed

variability is not associated with age of the donor or duration of occupational exposure. Smokers and non-smokers were also present in the studied groups. However, similarly to age, this factor did not affect the b/c values significantly. Those results are in agreement with reports of other authors [18,19]. It seems that exposure to tobacco smoke induced resistance to bleomycin. Probably, the cells acquired a mechanism to detoxify various toxic compounds [23]. It is important to recognise that the susceptibility to DNA damage varies in different individuals. Our results have also shown that some exposed subjects exhibited significantly higher level of DNA damage as expressed by means of chromatid breakage assay. It could be assume that subjects with a high susceptibility phenotype in our study may, in fact, be heterozygotes of the some chromosomal instability syndromes. Namely, heritability of persistent chromosomal damage is well established for the severe chromosomal instability syndromes such as ataxia teleangiectasia and Fanconi anemia [19]. Avoidance of harmful occupational exposure, especially in sensitive persons may become an important factor in the prevention of cancer.

The results of present study indicate that the alkaline comet assay, as sensitive biomarker of exposure, can be successfully applied in assessment of primary DNA damage in occupationally exposed subjects. As a microdosimetric technique based on the selection of individual cells in a heterogeneous cell population on the basis of nuclear morphology and shape of comets it is suitable for *in vivo* human biomonitoring. By gathering information on clastogenic and aneugenic impacts micronucleus assay is confirmed as sensitive biomarker of effect. On the other hand, chromatid breakage assay (bleomycin sensitivity test) seems to be a useful biomarker of susceptibility in predicting individual risk of cancer. All methods employed in the present study provide powerful tools for successful biomonitoring of populations occupationally exposed to microwave radiation.

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# THE ALKALINE COMET ASSAY AS A BIOMARKER OF PRIMARY DNA DAMAGE IN PERIPHERAL BLOOD LEUKOCYTES OF NUCLEAR MEDICINE PERSONNEL

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## **Abstract**

The aim of this study was to assess whether occupational exposure to chronic low doses of ionizing radiation in nuclear medicine departments may lead to genotoxicity. The alkaline comet assay was selected as a biomarker of exposure to evaluate the levels of primary DNA damage in peripheral blood leukocytes of exposed and corresponding control subjects. Statistically significant differences were found between comet tail length and tail moment values measured in leukocytes from the exposed and control groups. Within exposed group significant inter-individual differences in DNA damage were assessed, indicating different genome sensitivity. In majority of exposed subjects the levels of DNA damage were in positive correlation with the duration of occupational exposure, while the influences of age and dosimeter readings could be excluded. However, the levels of primary DNA damage detected both in control and exposed subjects were significantly influenced by smoking. The present study indicates the possibility of genotoxic risks related to occupational exposure in nuclear medicine departments. Therefore, the exposed personnel should carefully apply the radiation protection procedures to minimize, as low as possible, radiation exposure to avoid possible genotoxic effects. According to results obtained, the alkaline comet assay could be usefully applied as a sensitive additional biomarker in the regular health screening of workers occupationally exposed to low doses of ionizing radiation.

## **Introduction**

Ionizing radiation is an ubiquitous environmental physical agent whose DNA damaging effects are fairly well established. As a strong physical mutagen and a clastogenic agent, it causes the breakage of phosphodiester bonds in DNA, cross-linking as well as damage to nucleotide bases [1].

The exposure of patients and workers to radiation in medicine is a direct consequence of the use of radiation to improve the health of the individuals. Medical personnel employed in nuclear medicine departments is occupationally exposed to low doses of ionizing emissions from different radioactive isotopes such as  $^{32}\text{P}$ ,  $^{67}\text{Ga}$ ,  $^{111}\text{In}$ ,  $^{201}\text{Tl}$ ,  $^{59}\text{Fe}$ ,  $^{57}\text{Co}$ ,  $^{51}\text{Cr}$ ,  $^{99\text{m}}\text{Tc}$ ,  $^{131}\text{I}$ ,  $^{192}\text{Ir}$  [2,3]. Over the past two decades, medical occupational radiation exposure has decreased in spite of increased use of radiation in medicine. This decrease reflects primarily the introduction of new technologies in medicine. However, possible consequences from exposure to low radiation doses as in diagnostic radiology, occupational activities or in nuclear medicine investigations should be studied to determine potential deleterious genetic effects to exposed populations. Evaluations of the cytogenetic impact of chronic low dose radiation exposure are still scarce. It is very important to estimate absorbed doses from individuals occupationally exposed to ionizing radiation for carrying out radioprotection procedures and restrict the hazards to human health [4].

In order to obtain information concerning distribution and extent of radiation exposures, different biological methods for dose assessment have been developed. Human biomonitoring can be performed using various genetic markers which detect early biological effects. Among them, the most frequently used are evaluation of DNA mutations, chromosomal aberrations, the induction of micronuclei and sister chromatid exchanges. Under the many biological parameters studied, the most fully developed biological indicators of ionizing radiation exposure are unstable chromosomal aberrations (in particularly dicentrics) in peripheral blood lymphocytes [4,5]. This methodology usually complements data obtained by physical dosimetry. As a routine, it is used whenever the individual dosimeter shows an exposure to penetrating radiation above its limit of detection. One of the advantages of cytogenetic dosimetry is that this biological dosimeter can be assessed at any moment whereas physical dosimeters are not always present in the subject [4].

During the last decade, the single cell gel electrophoresis (SCGE) or comet assay was introduced as a useful technique for human biomonitoring studies [6,7]. While biomonitoring studies employing cytogenetic techniques are limited to circulating lymphocytes and involve proliferating cell populations, the comet assay can be applied to proliferating and non-proliferating cells and cells of those tissues which are the first sites of contact with genotoxic agents. In this assay, increased DNA damage is visualized on the individual cell level as an increased migration of genetic material from the nucleus in the direction of electrophoresis, and this displacement can be quantified using computerized image analysis. The comet assay is a rapid, simple and sensitive technique for measuring DNA breakage with a small number

of cells and detects intercellular differences in DNA damage [8,9]. Being sensitive, this assay seems particularly well suited for studying radiation-induced DNA damage. By using the comet assay positive results on the DNA-damaging effects of X-radiation on occupationally exposed subjects were reported previously [10].

The aim of the present study was to assess and quantificate the levels of primary DNA damage in peripheral blood leukocytes of medical workers employed in nuclear medicine departments and corresponding non-exposed control subjects. As a sensitive biomarker of exposure the alkaline comet assay was selected.

### **Subjects and method**

The population studied comprised 40 volunteer blood donors: 20 of them had been occupationally exposed to ionizing radiation and 20 were unexposed control subjects. Each subject completed a standardized questionnaire which included items concerning personal data (age, health status) and data on occupational exposure to ionizing radiation at the time of the study. The questionnaire also included items concerning extra occupational exposure to potential mutagenic hazards, such as smoking, alcohol and drug consumption, viral diseases, recent vaccinations, and radio diagnostic examinations. The exposed group consisted of 12 female and 8 male subjects employed in the nuclear medicine department. They are occupationally exposed to low doses of particular ionizing emissions from different radioactive isotopes ( $^{32}\text{P}$ ,  $^{67}\text{Ga}$ ,  $^{111}\text{In}$ ,  $^{201}\text{Tl}$ ,  $^{59}\text{Fe}$ ,  $^{57}\text{Co}$ ,  $^{51}\text{Cr}$ ,  $^{99\text{m}}\text{Tc}$ ,  $^{131}\text{I}$ ,  $^{192}\text{Ir}$ ). Three of them were physicians, 10 technicians, 2 engineers, 4 clerks and one cleaner. Their mean duration of occupational exposure was 15.5 years (range 1-35 years). The mean age of the whole exposed group was 40.9 (age range: 26-56 years). During their work they all wore individual dosimeters (film badges). Their dosimeter readings during one year prior the study were in range 0-970  $\mu\text{Sv}$ . Eight exposed subjects were smokers (5 female and 3 male), and 12 non-smokers (5 female and 7 male). Control subjects were healthy students and office employees (12 females and 8 males), chosen from the general Croatian population. Thirteen of them were non-smokers (7 female and 6 male), and seven of them were smokers (5 female and 2 male). The mean age of the control group was 40.9 years (age range: 26-56 years). None of them had ever had any contact with sources of ionizing radiation. They also had not been occupationally exposed to known genotoxic agents. None of the control subjects reported alcohol consumption. During the year prior to the blood sample collection, the control had not been subjected to diagnostic X-ray or nonionizing examinations. Peripheral blood samples of

the exposed and control subjects were collected by venipuncture into heparinised tubes. All samples were coded, cooled and processed within a maximum of 2 h after collection.

#### *The alkaline comet assay*

The comet assay was carried out under alkaline conditions, basically as described by Singh et al. [6]. Fully frosted slides were covered with 1 % normal melting point (NMP) agarose (Sigma). After solidification, the gel was scraped off from the slide. The slides were then coated with 0.6 % NMP agarose. When this layer had solidified a second layer containing the whole blood sample mixed with 0.5 % low melting point (LMP) agarose (Sigma) was placed on the slides. After 10 minutes of solidification on ice, slides were covered with 0.5 % LMP agarose. Afterwards the slides were immersed for 1 h in ice-cold freshly prepared lysis solution (2.5 M NaCl, 100 mM Na<sub>2</sub>EDTA, 10 mM Tris-HCl, 1 % Na-sarcosinate (Sigma), pH 10) with 1 % Triton X-100 (Sigma) and 10 % dimethyl sulfoxide (Kemika) added fresh to lyse cells and allow DNA unfolding. The slides were then placed on a horizontal gel-electrophoresis tank, facing the anode. The unit was filled with fresh electrophoretic buffer (300 mM NaOH, 1 mM Na<sub>2</sub>EDTA, pH 13.0) and the slides were set in this alkaline buffer for 20 min to allow DNA unwinding and expression of alkali-labile sites. Denaturation and electrophoresis were performed at 4°C under dim light. Electrophoresis was carried out for 20 min at 25 V (300 mA). After electrophoresis the slides were rinsed gently three times with a neutralisation buffer (0.4 M Tris-HCl, pH 7.5) to remove excess alkali and detergents. Each slide was stained with ethidium bromide (20 µg/ml) and covered with a coverslip. Slides were stored at 4°C in sealed boxes until analysis. A total of 100 randomly captured comets from each slide were examined using an epifluorescent microscope (Zeiss) connected through a black and white camera to an image analysis system (Comet Assay II, Perceptive Instruments Ltd., U.K.). A computerizing image analysis system acquires images, computes the integrated intensity profiles for each cell, estimates the comet cell components, and then evaluates the range of derived parameters. To quantify the DNA damage, the following comet parameters were evaluated: tail length (TL), and tail moment (TM). Tail length (length of DNA migration) is related directly to the DNA fragment size and presented in micrometers. It was calculated from the centre of the cell. Tail moment was calculated as the product of the tail length and the fraction of DNA in comet tail. The data on comet parameters measured in the exposed and control groups were logarithmically transformed and evaluated by using non-parametric Mann-Whitney *U* test Differences were considered to be statistically significant when  $p < 0.05$ .



## Results

Results of the alkaline comet assay have been summarized in Tables 1 and 2. It was found out that medical workers who were employed at nuclear medicine department for different periods of time showed a significant increase in DNA migration compared to the control values ( $p < 0.05$ , Mann-Whitney  $U$  test). Within the exposed group marked inter-individual differences regarding comet tail parameters were also observed (Table 1).

The comet tail lengths measured in exposed subjects were in the range  $18.17 \pm 0.40 \mu\text{m}$  to  $32.54 \pm 0.66 \mu\text{m}$ , with a mean value  $23.40 \pm 0.17 \mu\text{m}$  (Table 1). Their tail moments ranged between  $15.76 \pm 0.46$  and  $28.82 \pm 0.63$ , and the mean value was  $21.01 \pm 0.26$  (Table 1). In majority of exposed subjects duration of occupational exposure was positively correlated with the extent of DNA damage recorded. The age did not significantly affect the observed DNA damage levels. Similar was found out for dosimeter readings recorded in the majority of exposed subjects. Moreover, in some exposed subjects without any registered dosimeter readings significant increases of primary DNA damage levels were observed (Table 1). It should be pointed out that the levels of DNA damage were significantly elevated due to smoking status. Taken together, exposed smokers had significantly increased values of both comet parameters compared to exposed non-smokers ( $p < 0.05$ , Mann-Whitney  $U$  test, Table 1). Similar was found out for exposed female non-smokers compared to exposed male non-smokers.

Control subjects showed a low level of DNA damage in their leukocytes. The distribution of comet tail parameters in control group was characterized by nuclei with small comets (Table 2). Their comet tail lengths were in the range  $12.59 \pm 0.12 \mu\text{m}$  and  $15.60 \pm 0.14 \mu\text{m}$  with a mean value of  $14.28 \pm 0.04 \mu\text{m}$ , while the tail moment values were in the range  $10.01 \pm 0.13$  to  $13.09 \pm 0.15$ , with a mean value of  $11.77 \pm 0.04$  (Table 2). Unexposed smokers had significantly increased levels of DNA damage compared to non-smokers. It was also observed that the tail moment values in unexposed male subjects were significantly higher compared to female ( $p < 0.05$ , Mann-Whitney  $U$  test). However, similar was not observed for the corresponding tail lengths.

**Table 1.** The individual results of the alkaline comet assay on peripheral blood leukocytes of medical personnel employed in nuclear medicine department expressed as mean values the measurements of 100 comets per each subject.

Subject No.	Gender Smoking	Age (years)	Occupation	Exposure (years)	Dose ( $\mu\text{Sv}$ )*	Tail length ( $\mu\text{m}$ )			Tail moment		
						Mean $\pm$ SE	Min.	Max.	Mean $\pm$ SE	Min.	Max.
1	M <sup>N</sup>	36	technician	15	970	18.26 $\pm$ 0.46	12.18	40.38	15.76 $\pm$ 0.46	9.38	36.60
2	F <sup>S</sup>	31	technician	10	760	29.60 $\pm$ 0.73	14.74	51.28	26.33 $\pm$ 0.71	12.16	47.44
3	F <sup>N</sup>	45	clerk	23	660	26.66 $\pm$ 0.78	15.38	70.51	23.55 $\pm$ 0.75	12.28	66.30
4	M <sup>N</sup>	48	physician	17	500	23.42 $\pm$ 0.62	16.03	69.23	20.63 $\pm$ 0.61	13.41	65.39
5	M <sup>N</sup>	26	technician	7	340	18.17 $\pm$ 0.40	12.18	41.02	15.78 $\pm$ 0.40	9.70	37.77
6	M <sup>S</sup>	45	technician	24	330	18.87 $\pm$ 0.41	13.46	40.38	16.28 $\pm$ 0.38	10.90	34.97
7	F <sup>S</sup>	51	engineer	32	320	18.21 $\pm$ 0.31	12.82	29.69	15.78 $\pm$ 0.31	10.02	26.60
8	F <sup>S</sup>	30	technician	6	281	18.40 $\pm$ 0.34	13.46	35.26	15.90 $\pm$ 0.33	10.89	31.94
9	F <sup>N</sup>	43	technician	16	250	24.46 $\pm$ 1.20	12.18	107.05	21.40 $\pm$ 1.13	9.64	100.20
10	M <sup>S</sup>	56	physician	13	170	25.49 $\pm$ 0.85	15.38	75.00	22.42 $\pm$ 0.82	12.79	71.66
11	F <sup>N</sup>	46	cleaner	8	100	18.64 $\pm$ 0.31	14.10	29.49	15.85 $\pm$ 0.29	11.40	26.68
12	F <sup>S</sup>	45	clerk	9	0	32.54 $\pm$ 0.66	18.59	55.13	28.82 $\pm$ 0.63	15.49	48.53
13	F <sup>N</sup>	26	technician	8	0	28.97 $\pm$ 0.72	16.03	57.05	25.82 $\pm$ 0.71	12.99	52.91
14	M <sup>N</sup>	41	engineer	15	0	27.17 $\pm$ 0.62	17.95	45.51	23.98 $\pm$ 0.59	14.34	40.89
15	M <sup>S</sup>	54	physician	27	0	26.02 $\pm$ 0.72	13.46	66.02	22.85 $\pm$ 0.68	10.63	60.63
16	F <sup>N</sup>	53	clerk	35	0	25.86 $\pm$ 0.60	16.03	55.77	22.65 $\pm$ 0.58	13.09	51.89
17	F <sup>N</sup>	29	technician	7	0	23.03 $\pm$ 0.44	13.46	35.26	20.11 $\pm$ 0.42	10.92	31.86
18	F <sup>S</sup>	48	technician	28	0	22.54 $\pm$ 0.51	12.82	43.59	19.69 $\pm$ 0.49	9.90	38.51
19	F <sup>N</sup>	35	clerk	1	0	22.53 $\pm$ 0.42	15.38	42.31	19.48 $\pm$ 0.40	12.75	39.13
20	M <sup>N</sup>	31	technician	8	0	19.17 $\pm$ 0.48	12.18	41.02	16.58 $\pm$ 0.46	9.64	37.28
<b>Mean values <math>\pm</math> SE</b>											
<b>Smokers</b>						<b>23.96 <math>\pm</math> 0.28-</b>			<b>21.01 <math>\pm</math> 0.26-</b>		
<b>Non-smokers</b>						<b>23.03 <math>\pm</math> 0.21</b>			<b>20.13 <math>\pm</math> 0.20</b>		
<b>EXPOSED GROUP</b>						<b>23.40 <math>\pm</math> 0.17</b>			<b>20.48 <math>\pm</math> 0.16</b>		

M-male; F-female; S-smoker; N-non-smoker; \*cumulative dose measured in the last one year before blood sampling; -  $p < 0.05$  compared to control (Mann-Whitney  $U$  test)

**Table 2.** The individual results of the alkaline comet assay on peripheral blood leukocytes of control population expressed as mean values the measurements of 100 comets per each subject.

Subject No.	Gender Smoking	Age (yrs)	Tail length ( $\mu\text{m}$ )			Tail moment		
			Mean $\pm$ SE	Min.	Max.	Mean $\pm$ SE	Min.	Max.
1	M <sup>N</sup>	36	14.63 $\pm$ 0.17	10.26	19.23	12.10 $\pm$ 0.17	7.88	15.89
2	F <sup>S</sup>	31	14.55 $\pm$ 0.15	10.90	19.23	11.96 $\pm$ 0.15	7.91	16.43
3	F <sup>N</sup>	45	14.63 $\pm$ 0.17	10.26	17.95	12.14 $\pm$ 0.16	8.26	15.39
4	M <sup>N</sup>	48	14.00 $\pm$ 0.15	10.90	16.67	11.72 $\pm$ 0.14	8.57	14.47
5	M <sup>N</sup>	26	14.62 $\pm$ 0.16	10.26	17.95	12.05 $\pm$ 0.16	7.86	15.54
6	M <sup>S</sup>	46	15.03 $\pm$ 0.16	10.90	19.23	12.43 $\pm$ 0.16	8.56	16.71
7	F <sup>S</sup>	51	14.04 $\pm$ 0.14	10.90	17.95	11.45 $\pm$ 0.14	8.05	15.70
8	F <sup>S</sup>	30	15.60 $\pm$ 0.14	12.18	18.59	13.09 $\pm$ 0.15	9.47	16.07
9	F <sup>N</sup>	41	14.72 $\pm$ 0.14	11.54	17.95	12.13 $\pm$ 0.14	9.16	15.35
10	M <sup>N</sup>	56	13.81 $\pm$ 0.15	10.90	17.31	11.40 $\pm$ 0.14	8.39	14.47
11	F <sup>N</sup>	49	14.41 $\pm$ 0.14	11.54	17.95	11.80 $\pm$ 0.14	8.34	15.18
12	F <sup>S</sup>	45	13.37 $\pm$ 0.13	10.26	17.31	10.96 $\pm$ 0.12	7.98	15.14
13	F <sup>N</sup>	26	13.70 $\pm$ 0.13	11.67	16.86	11.09 $\pm$ 0.13	9.06	14.33
14	M <sup>N</sup>	41	14.04 $\pm$ 0.14	10.26	16.67	11.57 $\pm$ 0.15	7.47	14.71
15	M <sup>S</sup>	50	15.01 $\pm$ 0.18	10.26	18.59	12.56 $\pm$ 0.17	8.09	16.32
16	F <sup>N</sup>	53	14.17 $\pm$ 0.13	11.54	17.31	11.60 $\pm$ 0.13	8.70	14.68
17	F <sup>N</sup>	29	12.59 $\pm$ 0.12	10.37	13.61	10.01 $\pm$ 0.13	6.76	11.34
18	F <sup>S</sup>	48	13.81 $\pm$ 0.14	10.90	16.67	11.37 $\pm$ 0.14	8.39	14.14
19	F <sup>N</sup>	35	13.99 $\pm$ 0.15	10.26	16.67	11.60 $\pm$ 0.15	7.72	14.47
20	M <sup>N</sup>	31	13.76 $\pm$ 0.14	10.90	17.31	11.18 $\pm$ 0.13	8.20	14.66
<b>Mean values <math>\pm</math> SE</b>								
<b>Smokers</b>			<b>14.49 <math>\pm</math> 0.06-</b>			<b>11.98 <math>\pm</math> 0.06-</b>		
<b>Non-smokers</b>			<b>14.16 <math>\pm</math> 0.04</b>			<b>11.65 <math>\pm</math> 0.04</b>		
<b>CONTROL GROUP</b>			<b>14.28 <math>\pm</math> 0.04</b>			<b>11.77 <math>\pm</math> 0.04</b>		

M-male; F-female; S-smoker; N-non-smoker; -  $p < 0.05$  compared to control (Mann-Whitney  $U$  test).

## Discussion

Working in the presence of ionizing radiation is one of many managed risks within a hospital. However, over the past two decades, medical occupational radiation exposure has decreased in spite of increased use of radiation in medicine [11]. Occupational exposure in nuclear medicine departments is mainly related to low doses of particular ionizing emissions from radioactive isotopes such as  $^{32}\text{P}$ ,  $^{67}\text{Ga}$ ,  $^{111}\text{In}$ ,  $^{201}\text{Tl}$ ,  $^{59}\text{Fe}$ ,  $^{57}\text{Co}$ ,  $^{51}\text{Cr}$ ,  $^{99\text{m}}\text{Tc}$ ,  $^{131}\text{I}$ ,  $^{192}\text{Ir}$  [2,3]. These radioisotopes have unstable nuclei, and dissipate excess energy by spontaneously emitting radiation in the form of gamma and other rays. Increasing growth of nuclear medicine practice, the use of radioactive isotopes for diagnostic purposes (cell-labelling and tumor imaging) and the use of radiopharmaceuticals in therapy lead to an increase of potential genetic risks to medical personnel. Once a radiopharmaceutical has been administered, the patient becomes a mobile source of radiation exposure to several critical groups of individuals. This source of exposure consists of two distinct types: (i) exposure to photon radiation which is emitted by the radioactivity retained by the patient but has not been absorbed within the patient. (ii) contact with radioactive secretions, excretions or tissue from the patient [12]. Contrary to the patients, medical staff is usually exposed to much lower doses, but for a longer period of time.

In order to obtain information concerning the distribution and extent of radiation exposures, as such data are lacking from physical dosimetry in many cases, the use of biological indicators has gain increasing importance. Among individuals occupationally exposed to low levels of ionizing radiation different cytogenetic changes, for example increased frequencies of chromosome aberrations, micronuclei and SCE are usually evaluated [2-5,10,13]. An important advantage of biological dosimeters is the fact that they can be assessed at any moment, whereas physical dosimeters are not always present in the subject. Another advantage is that subjects under study cannot intentionally modify the biological dosimeter [2].

Lesions induced by ionizing radiation in DNA can be also detected by the alkaline single cell gel electrophoresis or comet assay [10,13]. This sensitive, quick and accurate test has been widely applied to measure both *in vitro* DNA damage and repair following exposure to various genotoxic agents and for human biomonitoring [7,8,13]. Results of the present study using the alkaline comet assay revealed an increase of radiation-induced primary DNA damage in human leukocytes of occupationally exposed nuclear medicine personnel.

Physico-chemical interactions between ionizing radiation and DNA produce a broad spectrum of DNA lesions including damage to nucleotide bases, DNA-DNA and DNA-protein cross-links, alkali labile sites as well as DNA single- and double-strand breaks (DSBs) [14]. Using the alkaline modification of comet assay we were able to detect DNA lesions induced, because the endpoint measured by traditional comet assay is a mixture of direct strand breaks and DNA damage that is converted to strand breaks by alkaline treatment (alkali-labile lesions) [7-9].

According to results obtained, the levels of DNA damage in majority of exposed subjects were positively correlated to the duration of occupational exposure. Observed heterogeneity in levels of DNA damage among exposed subjects could be addressed to specific modes of exposure as well as to different individual genome sensitivity. Because specialized workers often tend to perform the same tasks, it could be expected that some of them would exhibit higher levels of DNA damage. High levels of DNA damage recorded in leukocytes of occupationally exposed medical personnel suggest that working in area with continuous low level exposure may result in detectable primary DNA damage in circulating leukocytes caused either or/both by occupational contamination or by different clastogenic factors. These factors are responsible for the delayed effects of radiation. Among them there are superoxide radicals, that may induce genotoxic effects by indirect-acting mechanisms [15]. It could also be taking into account that nuclear medicine personnel is exposed to different radionuclides, that are characterized by different half-life and different emissions. Furthermore, some radionuclides are taken up in different amounts by different organs that might act as internal sources of irradiation. One of example is  $^{131}\text{I}$  with an effective half-life of 8 and 3 days that might accumulate in thyroid gland [16]. It can be considered that nuclear medicine personnel is chronically exposed to small doses of different radionuclides which continuously causing lesions in DNA. Therefore, increased levels of primary DNA damage recorded by comet assay could be also related to accumulation of non-repaired DNA damage.

The accumulated dose range in nuclear medicine workers involved in our study was from 0  $\mu\text{Sv}$  to 970  $\mu\text{Sv}$ . However, we did not find a clear correlation between the DNA damaging effects and the doses recorded by the dosimeters. This result is in agreement with studies mentioned previously [10,17]. The authors discussed that it is difficult to establish dose-effect relationships for the low doses. It is also possible that certain parts of the body of the same subjects were nonhomogeneously exposed to higher radiation doses than those encompassed by the control of the dosimeter.

The comet assay allows determination of primary DNA damage, providing an assessment of current exposure. In addition to radionuclide exposure, other factors may contribute to enhanced levels of DNA damage, such as lifestyle associated factors. Our study confirms previous reports on smoking as a confounding factor that strongly influenced the basal levels of primary DNA damage both in exposed and control subjects. However, it has to be pointed out that the age of the individual had no significant effects on the mean basal level of DNA damage neither in exposed nor in control subjects of our study. These observations are also in agreement with previous reports of other authors [8,19].

Comet measurements may reflect both individual repair ability and DNA damage level. Because the measured damage level is a result of equilibrium between damage infliction and repair, a low damage level as assessed experimentally in an individual may be the result of an actual low number of lesions or of a high efficiency of repair. DNA damage detected with the comet assay is quickly repaired, usually within 2 h [20]. Considering that the DNA damage visualized by the comet assay is not permanent, high levels of DNA damage in exposed subjects could be ascribed to a general negative effect of exposure, such as an increased susceptibility to the activity of ionizing radiation or a decreased efficiency in recognition and repair of induced damage. The detection of damage would also suggest the presence of continuously active agents such as clastogenic factors in plasma.

It is well known that the some types of DNA damage (especially chromosome aberrations) in peripheral lymphocytes of the exposed subjects are accumulated during the long period of occupational exposure. The accumulation may be related to the populations of long-lived peripheral lymphocytes.

In the case of chronic exposure, a cumulative effect of ionizing radiation become more important and in these circumstances cytogenetic biomonitoring may provide different information on the DNA damage levels compared to comet assay. Because comet assay detects momentary DNA damage and/or repair activity, it reflects the current exposure and the actual levels of DNA damage present in peripheral blood leukocytes of the nuclear medicine workers at the moment of blood sampling.

Exposure to X-rays may result in effects that are both deterministic and stochastic in nature. Radiation workers are only at increased risk for deterministic effects if they work in violation of hospital safety policy and procedure [11]. Some stochastic effects to occupational exposure to X-rays would be cancer and genetic mutation. These are long term effects, which do not show in the exposed population until many years after the exposure. Under any circumstance, it is not acceptable for nuclear medicine workers to exceed their annual limit. It

is well documented that modern equipment and moderately aggressive radiation protection practices can keep occupational exposures at least an order of magnitude below the annual limit for the vast majority of medical workers. Rotation of workers through all tasks would also tend to distribute occupational exposures relatively uniformly over time.

In short, our study demonstrates the presence of increased primary DNA damage in nuclear medicine personnel exposed to low doses of ionizing radiation. Therefore, the personnel who work in nuclear medicine departments should carefully apply the radiation protection measures to minimize, as low as possible, radiation exposure to avoid possible deleterious genotoxic effects. The reported data also indicate that the standard alkaline comet assay protocol may be suitable for the biomonitoring purposes, as an additional complement to the standard biological dosimetry methods.

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# CONSTRUCTION AND APPLICATION OF $^{210}\text{Po}$ ALPHA-EMITTING EXPOSURE DEVICES FOR RADIOBIOLOGICAL EXPERIMENTS

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## **Introduction**

One way of studying the risk of low-level ionising radiation exposure to human health is to carry out biological experiments on living cell cultures. The general objective of the present work is to contribute to the exploration of the relationship between biological effects and radiation burden caused by high LET radiation investigating the correlations between factual hits or macroscopic dose and biological responses. The specific objectives are to construct appropriate alpha-irradiating systems and to study the targeted and non-targeted effects of alpha particles in human lung cell cultures to quantify the biophysical effect of radon and its progenies.

## **Materials and methods**

A  $^{210}\text{Po}$  alpha-exposure device has been constructed with a set of alpha sources, which differ in their activity: 38.4, 3.5 and 0.4 MBq (2003.06.11.). The  $^{210}\text{Po}$  isotope was used, because it is monoenergetic (5.297 MeV) and practically pure alpha particle emitter. It was electroless deposited on the surface of silver discs of 110 mm diameter and 0.2 mm thickness, with a useful diameter of 90 mm. The sources were fixed in aluminium holders and marked: 100-1, 100-2, 100-3. Calculations were made to decide the length of the collimator, which has cylindrical channels of 2 mm diameter. In Fig. 1, the results with 2 and 4 mm thick collimators are presented: 4 mm would make the alpha-beam narrower, but it would decrease the intensity in a high percentage, therefore 2 mm was chosen.

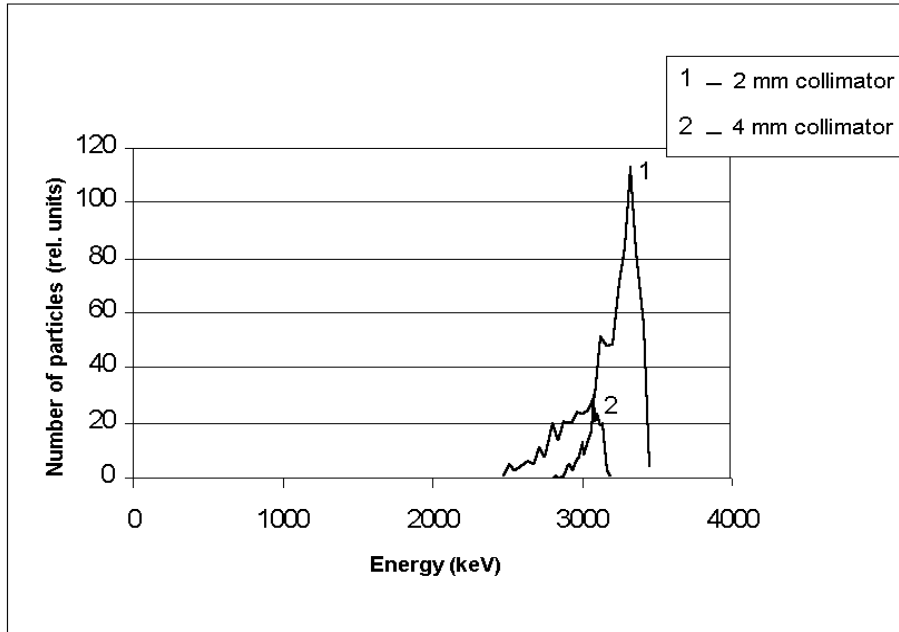


Fig. 1. SRIM 2003 calculations.

The nominal diameter of the collimator is 90 mm, and its truss covers 17% of the useful surface. Beyond and above it Mylar foils of 2.2  $\mu\text{m}$  thickness are placed to avoid the accumulation of dust and other agents on the source.

The activity distribution on the surface of the sources is not perfectly homogeneous; the inhomogeneities are between 18-22%. Therefore, an electronic source-rotating device was developed to decrease this non-uniformity and the one, which originates from the collimator arrangement. The device, in which the sources can be changed, is shown in Fig. 2.



Fig. 2. The source-rotating device with Petri dishes.

The cells are irradiated in 6  $\mu\text{m}$  Mylar based stainless steel Petri dishes placed above the collimated rotating source by shifting a horizontal shutter. The Petri dishes are also presented in Fig. 2 at the bottom of the photo. The required frequency of the source rotation is about 10-50 revs/min and can be fixed by the aid of a potentiometer (left side of Fig. 2).

Estimates of dose rate at the cell surfaces were received by measurements with a PIPS detector and by calculations with the SRIM 2003 software, which is a Monte Carlo charged particle transport code.

## Results

Inspection of the results demonstrates that 82% of the alpha particles hitting the cells have energies between 2.8 and 3.2 MeV and traverse the cell monolayer with an average energy loss of 1.589 MeV. Some data of the sources are summarised in Table 1.

Table 1. Data of the alpha sources (11.06.2003.)

<b>Mark of the source</b>	100-1	100-2	100-3
<b>Average alpha flux (alpha/(cm<sup>2</sup> s))</b>	14156	1698	189
<b>Inhomogeneity (%)</b>	15	7,5	20
<b>Hit probability (alpha/(cell 100 s))</b>	0,71	0,085	0,0095
<b>Dose rate (mGy/s)</b>	4,5	0,54	0,06

The calculations were carried out for the geometry of the source-rotating device and for particles which traverse an 8  $\mu\text{m}$  cell-equivalent A 150 phantom material layer. The average energy loss is 1.589 MeV.

In Fig. 3, the spectra taken by a PIPS alpha spectrometer are presented in different geometrical arrangements in the case of the 100-1 source for a single channel of the collimator. The 4 mm thick collimator makes the alpha beam narrower than the 2 mm thick, but it decreases the fluence rate in a high percentage. The additional Mylar foils broaden the spectrum and decrease the mean energy and consequently the position of the maximum energy, and thus, the

energy of the alpha particles hitting the cells can be varied in the required way. During the present experiments an arrangement corresponding to the 3<sup>rd</sup> spectrum is used.

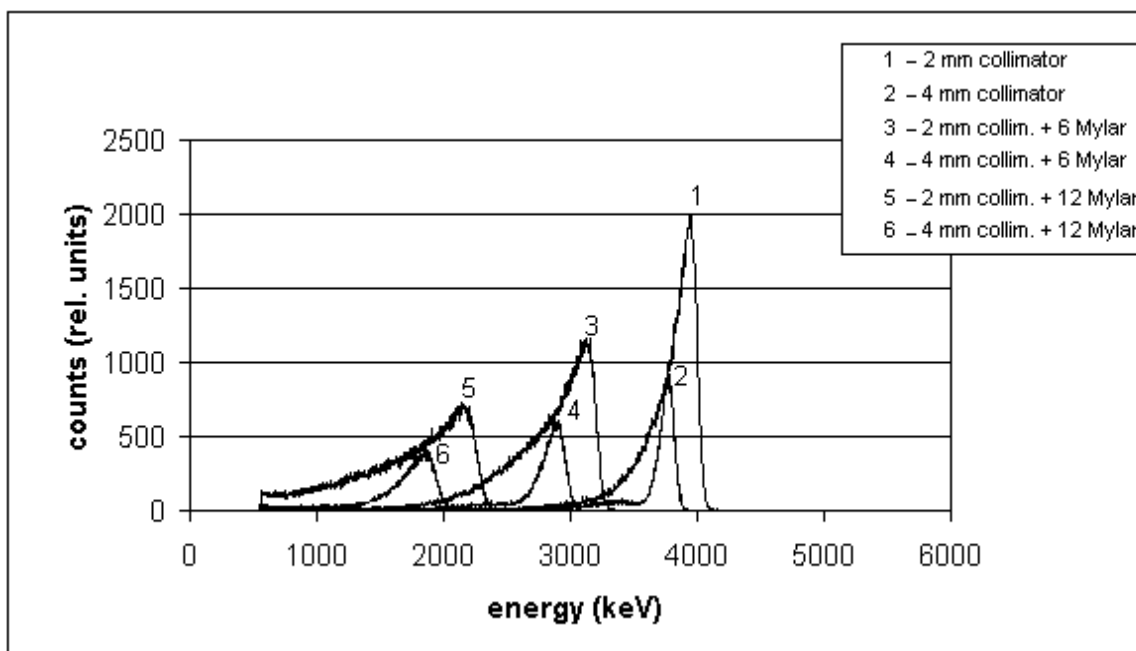


Fig. 3. Spectra taken by PIPS detector.

Several types of cell biological experiments are being carried out with this alpha-irradiating system e.g. investigation of adaptive response, bystander effect and their correlations, measurements of transformation probabilities as a function of alpha dose and alpha-hit probabilities; genomic instability studies for radiation induced mutations and neoplastic transformations.

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# THE ROLE AND IMPACT OF REFERENCE DOSES ON DIAGNOSTIC RADIOLOGY, HOW TO USE THEM AT THE NATIONAL LEVEL?

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## **Introduction**

In view of the fact that patient doses for similar types of diagnostic examinations cover ranges of as much as two orders of magnitude, dose reference levels (DRL) were proposed by the European Commission [1], with the intention to serve as a practical tool for promoting optimisation of the patient protection. The well known objective of the optimisation is to produce diagnostic images of adequate quality at the lowest possible radiation dose to patient and using a real QA programme to find the optimal balance between this two key parameters: image quality and doses. The doses obtained during diagnostic procedures are expressed in terms of the directly measurable dose quantities, it means: entrance surface dose (ESD) per radiograph, dose-area product (DAP) per examination, or average glandular dose (AGD) in mammography.

The generally recommended DRL values for common used typical examinations of standard sized patients and broadly defined types of equipment were set up mostly using national surveys in most developed countries.

Officially it is supposed that if reference dose levels are exceeded in some diagnostic procedures, there should be a local review of the procedures and equipment parameters, with the aim to take measures for dose reduction.

As a pragmatic approach third quartile values were adopted for identification of 25% of X-ray departments, for each type of examination obtained from the distribution of doses to a group of standard patients.

The usefulness and benefit of the DRL implementation into the national legislation is without any doubt a very important process of patient dose reduction. As the open question of the DRL concept still remains the fact that the observed dose distributions are specific to one

country or region and equal possibilities to compare the real obtained doses with the recommended EC DRL values, actually need establishing of an action plan for national QA programme and realisation of above mentioned dose surveys, which may set up DRL standards lower than are published, or where circumstances dictate, slightly higher DRL could be justified. In the paper are presented results of patient dose measurements studies realised in Slovak republic, in the framework of activities of the Slovak Commission of Quality Assurance established by the Slovak Ministry of Health.

### **Material and Methods**

The information of doses obtained during radiographic practices has been collected for:

PA chest examinations in paediatric radiology [2] and chest examinations of adult patients using radiophotography and radiography examination

Estimation of absorbed doses to embryo from urography examinations of pregnant woman

Assessment of doses to patients submitted during the fluoroscopic gastrointestinal tract examinations

Measurement of ESD and AGD on chosen slovak mammography units, involved in the national QA/QC audit

For the realisation of the dose measurement TLD dosimeters LiF 700, Harshaw were used, DAP meter and special phantoms:

Tissue equivalent phantom of the standard adult men (Alderson)

Tissue equivalent phantom of a child (about 1 year old)

Standard mammography phantom RMI 156

Specially prepared phantom simulating the woman in the first trimester of pregnancy

### **Results**

In all presented examples are given the minimal and maximal measured values and the mean doses. Where possible also the values of third quartil are added.

Estimation of chest examination doses on childrens was realised at 9 hospitals where radiological examinations of children are performed. Recommended reference value of ESD for chest examination in PA projection is 100  $\mu\text{Gy}$  [3]. Results are shown in the table 1.

Tab. 1 Entrance Surface Dose of 0-15 year old children for the chest PA/AP examination

Age Category	ESD ( $\mu\text{Gy}$ )					
	min	max	median	1st quartile	mean	3rd quartile
0-15 years	222	788	427	356	427	507
0-4 years	222	623	508	347	422	483
5-9 years	226	788	367	359	416	541
10-15 years	257	754	-	368	444	548

Estimation of the dose - area products during the radiologic examinations of small intestine were performed at 3 radiological departments. The results are given in the table 2. The dose reference level (DRL) for small intestine examination is  $25 \text{ Gy}\text{cm}^2$  [3, 4].

Tab.2 Summary of the results for small intestine

Type of examination	No. of patient	DAP ( $\text{Gy}\text{cm}^2$ )					
		min	1st quartile	mean	median	3rd quartile	max
enteroclysis (C)	14	7,9	18,2	24	21,5	32,4	44,6
enteroclysis (D)	5	6,4	15,7	22,1	18,1	18,3	52,0
barium follow-through procedures (C)	24	2,9	11,2	13,3	12,2	15,2	26,7
barium follow-through procedures (D)	25	2,4	5,0	10,1	6,5	9,3	39,5

C – conventional  
D – digital

The study of the absorbed doses to embryo during IVU examinations was performed on 14 radiography units with results which can be seen in the table 3. Recommended reference value of ESD for IVU examination is 10 mGy and critical dose to embryo is 50 mGy during whole time of pregnancy [3, 5].

Tab.3 Entrance Surface Dose (for 1 exposure) and Absorbed Dose (for 3 exposures) to Embryo measured during IVU examinations

	min	1st quartile	mean	median	3rd quartile	max
Entrance Surface Dose (mGy)	3,5	9,45	15,5	14,1	19,15	36,2
Absorbed Dose to embryo (mGy)	5,1	13,1	19	18,7	23,15	34,4

The chest PA examinations were performed on 14 radiological departments, where simultaneously radiography and radiophotography units are working. Results of the study are

in the table 4. Recommended reference value of ESD for radiography examination of chest in PA projection is 0,4 mGy [2, 3].

Tab. 4 Entrance Surface Dose for the chest PA examination

ESD (mGy)	min	1st quartile	mean	median	3rd quartile	max
radiophotography	3,7	9,9	14,2	15,1	18,7	20,5
radiography	0,107	0,27	0,546	0,416	0,499	2,498

The national audit of mammography units is realised on 21 mammography departments, the entrance surface doses are given in table 5. Recommended reference value of ESD for mammography is 7 mGy [3].

Tab. 5 Entrance Surface Dose for mammographic examinations

ESD (mGy)	min	1st quartile	mean	median	3rd quartile	max
mammography	1,628	4,358	5,790	5,74	6,94	12,76

## Conclusions

Results of patient dose audits reported in our paper for several types of examinations and various technical units have shown the importance of applications of reference dose levels in radiological practice. On the basis of national surveys slightly lower or higher standard DRL values could be justified. Continuing revision of DRL values and their extension to other types of radiographic and fluoroscopic examinations is needed.

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# THE PATIENT EXPOSURE FROM CHEST EXAMINATION IN DIAGNOSTIC RADIOLOGY

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## **Introduction**

The chest examination remains the most frequently applied radiological investigation. The medical exposure should be justified by weighing the benefit brought to healthy state against the radiation detriment which is represented by radiation doses to the patient. The diagnostic quality of radiological image and radiation dose to the patient depend on the most important technical parameters of radiological systems which must be continuously tested as part of quality control activities.

## **Material and methods**

The survey was carried out in the three representative units in Cluj city for chest examination (two hospitals and one ambulatory centre). The purpose of the study is to verify the most technical parameters of radiological system which influence the image quality and patient exposures.

The equipment used are an electronic device (multi-function meter RMI 242A) which measures the waveform, kVp and the time exposure and a dosimeter (RADCHECK Plus) for measurements of output radiation of diagnostic X-ray machine. The measurements were made for a wide range of values selected on control panel and the recorded values were compared with suggested acceptance limits (IAEA). The effective doses were derived from entrance surface dose (ESD) measured for typical exposure condition for PA projection chest examination, using PCXMC software. The number of expected fatal malignancies was

calculated as the product of the effective dose per examination, the number of examinations and the normalised induced malignancy probability.

## Results and discussion

The results of the quality control performed for technical parameters of radiographic system show the following:

The kilovoltage peak (kVp) is generally reproducible (stay the same exposure to exposure); exception is Philips X-ray system (table 1).

**Table 1. kVp reproductibility**

Radiological units	Radiological system	Physical parameters setup				Measured kVp	CV %	Permitted %
		FSD (cm)	mAs	ms	kV			
Pulmonary Tuberculosis Hospital	Philips Duodiagnost (CP)	75	20	82.3	70	68.9	0.222	<b>2</b>
						68.6		
						68.8		
Med I Hospital	Philips (1P-H)	75	20	60	70	68.8	2.74	<b>2</b>
						72.4		
						69.4		
Diagnostic and Treatment Centre-Adults	Eltex (3P-12)	100	23	80	70	45.3	0.55	<b>2</b>
						45.5		
						45.8		

Concerning kVp and time exposure (t) accuracy, the difference between the set and observed values of kVp and t exceeds permitted error (5%) for Eltex X-ray system at all (kVp,t) settings. For Philips X-ray system the kVp accurate, while exposure time has strong fluctuation (table 2).

To check radiation output linearity (mGy/mAs remains constant as the mA is varied) we found the maximum and minimum values of dose/mAs and calculated the coefficient  $k = \frac{\text{max} - \text{min}}{\text{max} + \text{min}}$ . Value of this coefficient exceeds permitted value only for Eltex X-ray system (table 3).

**Table 2. kVp and exposure time accuracy**

Radiological system	kVp				Exposure time			
	Set kV	Measured kV	Error %	Permitted %	Set ms	Measured ms	Error %	Permitted %
Philips Duodiagnost	60	58.9	1.86	5	78.5	78.5	0.00	5
	70	68.8	1.74		82.3	83.3	1.20	
	81	79.3	2.14		95.3	95.7	0.42	
	102	100.4	1.59		123	123.5	0.40	
	125	122	2.46		155.6	154	1.04	
Philips	60	59.1	1.52	5	50	51.3	2.53	5
	70	72.4	3.31		60	72.7	17.47	
	80	78.6	1.78		60	72.7	17.47	
	100	101.7	1.67		80	84.2	4.99	
	120	114.3	4.98		100	94.6	5.71	
Eltex	60	45.8	31.004	5	80	72.2	9.75	5
	70	45.5	53.846		80	72.2	9.75	
	80	51.1	56.555		80	72.4	9.50	
	100	70.9	41.043		80	74.1	7.37	

**Table 3. Radiation output linearity**

Radiological units	Radiological system	mAs set	Exposure in air mGy	Output mGy/mAs	k	Permitted
Pulmonary Tuberculosis Hospital	Philips Duodiagnost	50	2.41	0.0482	0.00233	0.1
		63	3.03	0.0481		
		80	3.86	0.0482		
		125	6.04	0.0483		
Med I Hospital	Philips	20	1.56	0.078	0.094	0.1
		40	3.48	0.087		
		60	5.66	0.0943		
		80	6.92	0.0865		
Diagnostic and Treatment Centre-Adults	Eltex	23	0.12	0.005	0.446	0.1
		40	0.32	0.008		
		62	0.7	0.011		
		102	1.39	0.013		

To check whether there is sufficient filtration in the X-ray beam to remove damaging low energy radiation we calculated half value layer (HVL) using Radcheck Plus to measure dose and a set of high purity aluminium layers. The value of HVL (thickness of aluminium which reduce the unattenuated dose by 50%) is unacceptable only for Eltex X-ray machine, relative to theoretical HVL of 2.64 mmAl at 80 kVp and 2.5 mmAl total filtration.

**Table 4. Half value layer (HVL) measurement**

Radiological system	Physical parameters setup				X ray beam quality		
	FSD (cm)	kVp	mAs	ms	HVL1 (mmAl)	HVL2 (mmAl)	Homogeneity factor
Philips Duodiagnost	75	81	50	-	2.77+/-0.07	7.43+/-0.24	0.37+/-0.01
Philips	75	80	50	0.16	2.93+/-0.04	7.19+/-0.14	0.41+/-0.01
Eltex	100	80	51	0.25	2.05+/-0.01	4.42+/-0.13	0.46+/-0.01

The table 5 show the values of entrance surface dose (ESD), with backscatter, and effective doses that patient receive from chest examination, in comparison with diagnostic reference level (DRL), ICRP report 73, 1996.

**Table 5. Patient exposure from chest examination**

Radiological system	Physical parameters setup for chest/PA examination				ESD* mGy	Ratio ESD /DRL	Effective dose (D) mSv	Ratio D /DRL
	FSD (cm)	kV	mAs	ms				
Philips	130	70	45	120	3.049	7.623	0.187+/-0.001	9.35
PhilipsDuodiagnost	80	70	125	681	13.876	34.69	0.817+/-0.007	40.85
Eltex	80	70	125	800	4.645	11.61	0.126+/-0.001	6.3

Statistically, chest X-ray examination remains a part of many routine health physical and screening programs aimed on finding heart and breathing diseases. The total chest X-ray exposures made on 2000 represents 32.6% from all types of X-ray procedures applied in Cluj city.

The number of expected fatal malignancies was calculated as the product of the effective dose per examination, the number of examinations and the normalised induced malignancy probability. The values found for hospitals and ambulatory centre is shows in table 6.

**Table 6. The associated fatal malignancies and detriment**

<b>Radiological units</b>	<b>Number of chest radiographies/2000</b>	<b>Number of expected fatal malignancies</b>	<b>Detriment</b>
Hospitals	4136	0.12	0.17
Ambulatory centre	2816	0.02	0.03

### **Conclusions**

Values found for ESD and effective doses are above diagnostic reference level in all thre units included in our study, especially because the high value of mAs. Despite the same protocols applied for PhilipsDuodiagnost and Eltex X-ray system, the variation in doses is due to the inaccurate kVp. This high values of ratio patient effective dose /guidance level require an investigation and optimisation of X-ray system or protocols.

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2. ICRP 73, Radiological Protection and Safety in Medicine, Annals of the ICRP, 26, 1996.
3. International Basic Safety Standards for Protection Against Ionizing Radiation and for the Safety of Radiation Sources, Safety Series115, IAEA, 1996.

# **APPROACH FOR REDUCTION OF LATE EFFECTS CAUSED BY RADIATION THERAPY ON NORMAL TISSUE FOR PATIENTS OF PROSTATE CANCER**

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## ***Introduction***

Prostate cancer is one of the most frequently serious diseases among elderly male patients. This malignant disease can be treated by surgical intervention, chemotherapy or radiotherapy. Approximately every second patient is applied to radiotherapy during his treatment.

Radiotherapy uses permanent radioactive seeds in case of brachytherapy or X-rays up to 20 MV in case of a tele-therapeutic approach, where the dose to the tumor is delivered by a certain amount of fields through normal tissue or organs at risk, like the bladder, the rectum, the wall of the rectum or the hip joints. In order to spare unwanted dose in the normal tissue great efforts have been done by conformal radiotherapy. Nevertheless dose to the normal tissue or organs at risk can be reduced by particle therapy, like proton therapy, which convinces with high dose homogeneity in the tumor-volume and low dose deposition to the normal tissue.

## ***Material and Method***

Heavy charged particles provide superior macro-dosimetric properties. Charged or uncharged particles have micro-dosimetric characteristics that causes high linear energy transfer (LET). Protons and helium ions offer superior macro-dosimetric properties, as well as micro-dosimetric characteristics resulting in low LET.

The treatment planning program OptiRad™, which was developed by the Loma Linda University Medical Center (California/USA), gives the opportunity to compare photon

therapy, which is used daily in many medical centers, with protons, who are accelerated up to 250 MeV to a pre-calculated penetration depth in the synchrotron of the Loma Linda Department of Proton Therapy<sup>1</sup>. Using beam delivery systems, which provide photons or protons to the planning target volume from any direction, it is possible to compare different application scenarios. In this study typical clinical photon and proton approaches are compared, like a 4-field box, but also 2-field and 6-field techniques, which are characteristic for individual cases (Fig. 1)

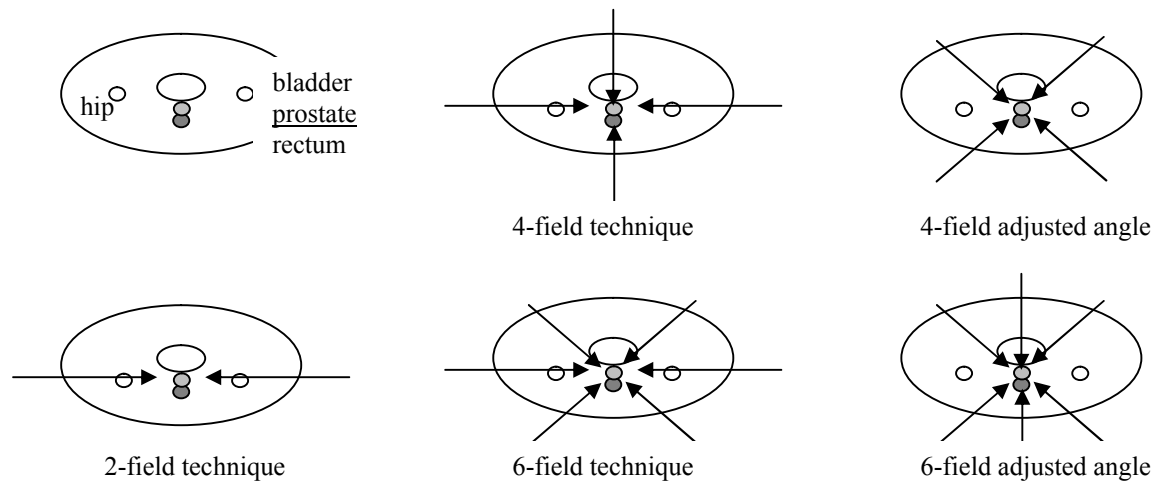


Fig. 1: different approaches for the radio-therapeutic prostate treatment (schematic), arrows indicate the beam direction for used techniques; in some cases it is necessary to adjust angles because of medical necessity, like metal hip joints.

In order to be able to compare different cases, the prescribed dose is unified to 72 Gy (CGE) while photon and proton treatment planning. All patients get fractionated therapy (1.8 Gy / CGE) per day resulting in an overall treatment time of 40 days. Photon techniques are calculated for the characteristics of a Siemens KD2, the proton techniques are elaborated for the Loma Linda synchrotron, gantry 1. In case of photon therapy individual blocks and wedges have been introduced when necessary. Individual beam weighting factors have been used in order to optimize the coverage of the tumor while reducing dose to the normal tissue. In the cases under consideration we discuss bladder, rectum, wall of the rectum and hips as organs at risk.

## **Results**

By means of 6 prostate cancer patients the dose distribution of the tumor and the normal tissue is discussed in a treatment planning comparison study of protons and photons.

Output factors like minimum dose to the tumor, maximum dose to the normal tissue or organ at risk, isodose – lines, integral dose and dose volume histograms (DVH) point up the advantages of particle radiotherapy. As an example for the benefit of particle the therapy dose to the planning target volume (PTV) is compared for the mentioned techniques by DVHs.

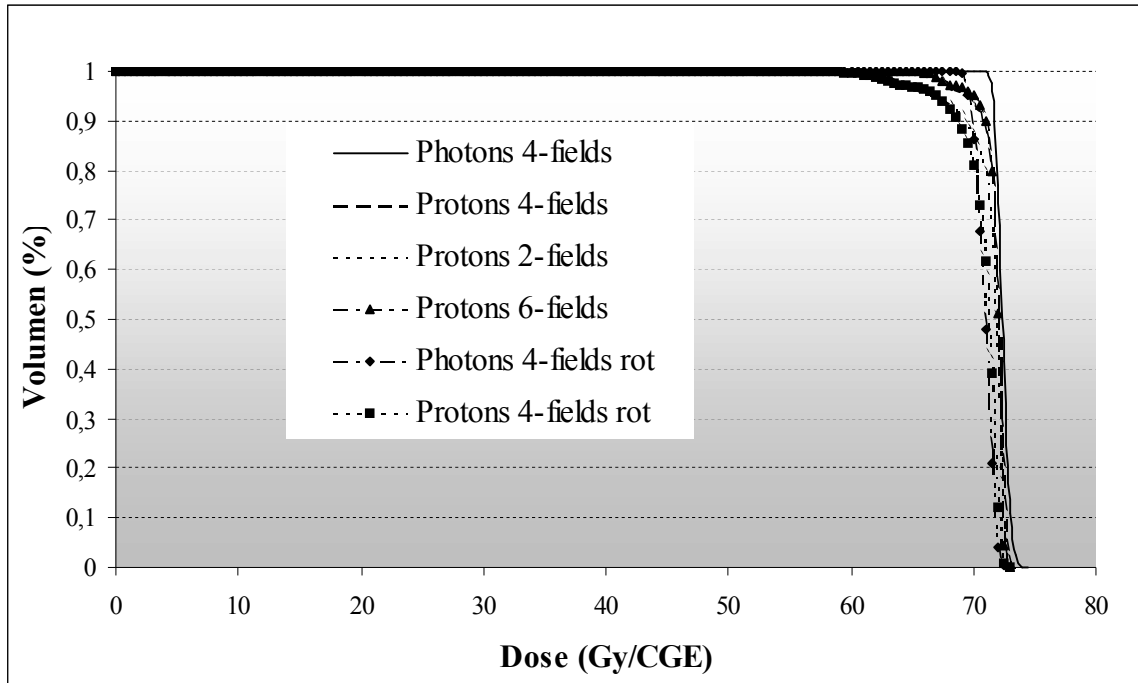


Fig. 2: Dose coverage of the prostate for individual techniques (rot = rotated i.e.: individually angulated); dose volume histogram for the PTV.

In all commonly used techniques the PTV is covered by a satisfactory way, which can be seen by the mean dose (table 1).

mean dose (Gy/CGE)	Patient 1	Patient 2	Patient 3	Patient 4	Patient 5	Patient 6
photons 4-fields	71.4	72.2	72.8	71.9	72.4	71.4
protons 4-fields	72.1	72.3	73.0	72.0	71.8	72.0
protons 2-fields	71.2	72.2	71.0	71.3	71.2	71.1
protons 6-fields	71.4	71.6	72.7	71.8	71.7	71.4
photons 4-fields rot	71.4	70.8	72.5	72.2	70.9	71.5
protons 6-fields rot	72.1	72.0	70.8	71.8	70.7	71.2

Table 1: mean dose to the tumor for 6 patients

In order to provide information about the dose to organs at risk or normal tissue DVH can assist the physician with questions of different techniques. As an example the DVH of the wall of the rectum is discussed.



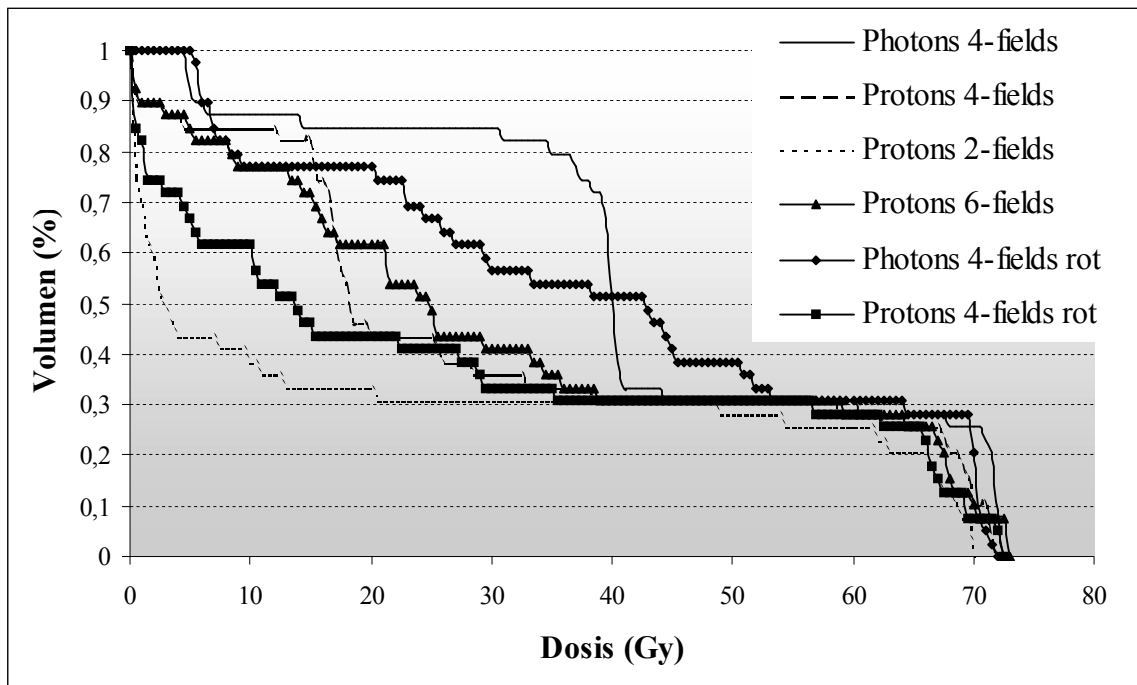


Fig. 3: DVH of the wall of the rectum

With respect to the different treatment techniques the wall of the rectum will get a significant higher dose, while using a photon 4-fields technique than a 2-field or 4-field proton technique.

### ***Discussion***

In order to increase tumor control probability, some multi-field dose escalation studies are in progress<sup>2</sup>. Treatment planning comparisons can help to decrease side- and late- effects by the calculation of the tumor control probability and the normal tissue complication probability. In any case, predicting the outcome of cancer treatment implies the knowledge of many specific parameters, like the tolerance dose ( $TD_{5/5}$ ,  $TD_{50/5}$ ) to the normal tissue or organ at risk. Nevertheless proton therapy is limited to the appropriateness of available particle therapy facilities, but can support advanced prostate cancer therapy by better tumor coverage and less side- and late- effects.

<sup>1</sup> Slater JM, Miller DW, Archambeau JO., Development of a hospital-based proton beam treatment center, Int J Radiat Oncol Biol Phys. 1988 Apr;14(4):761-75.

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# **Determination of the Biodistribution and Biokinetics of Radiopharmaca like $^{166}\text{Ho}$ -Ferric-Hydroxide or $^{153}\text{Sm}$ -EDTMP used for Therapeutic Treatment by Energy Dispersive Measurements**

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## **Introduction**

The activity distribution of beta-emitting radionuclides in the human body and the respective therapeutic dose distribution in the target and the unwanted leakage in the other organs was determined by measurement of corresponding gamma-lines. The measurement was done by scanning in a whole-body counter in the General Hospital Vienna. It is possible to localize activity and dose distribution by means of the detected activity profiles of the four detectors.

Two typical treatments are reported: the treatment of synovitis using radiation of  $^{166}\text{Ho}$ -Ferric-Hydroxide (characteristic gamma-line: 81 keV) and radionuclide therapy focused at the palliative treatment of bone metastases with  $^{153}\text{Sm}$ -EDTMP, a bone seeking beta-emitting radionuclide (characteristic gamma-line: 103 keV).

## **Materials and Method**

### *The clinical whole-body counter*

Whole-body counters are devices for the identification and measurement of small quantities of gamma-emitting radionuclides and their distribution in the human body. The Department of Nuclear Medicine of the General Hospital Vienna is equipped with a clinical whole-body counter, which was constructed by the Austrian Research Center in Seibersdorf.

This device consists of four 6“ x 4“ NaI(Tl) crystals, which are mounted in fixed positions: two stationary detectors above the couch and two below. The shadow shield is made of specially selected low background lead (length: 170 cm, thickness: 10 cm), which results in a total mass of 8500 kg. It is connected to a motor-driven, dead time corrected computer controlled scanning system during counting. During the measurements the computer-operated, variable-speed motor drives the couch for a preset scan time and scan length along an axis parallel to the longitudinal axis of the subject from the home-position through the measuring range to the end-position. The scan time was selected to 1000 s. The clinical whole-body counter is characterized by outstanding scan qualities due to precise adjustable slit collimators. Collimators can be adjusted between 0 mm and 300 mm in front of the detectors. Therefore this instrument is especially used for metabolic studies with patients, for research applications like the localization of small activities in the human body and for studies of the behaviour of radiopharmaceuticals, as well as for routine measurements.

### *Therapy*

A commonly used agent for the treatment of persistent synovitis is  $^{166}\text{Ho}$ -FH (characteristics see table 1). The activity of 1 GBq is administered in the knee. The emission of gammaradiation with an energy of 81 keV is used to monitor the kinetic distribution of  $^{166}\text{Ho}$  by performing activity profiles with the whole-body counter on 1, 3 and 7 days after therapy .

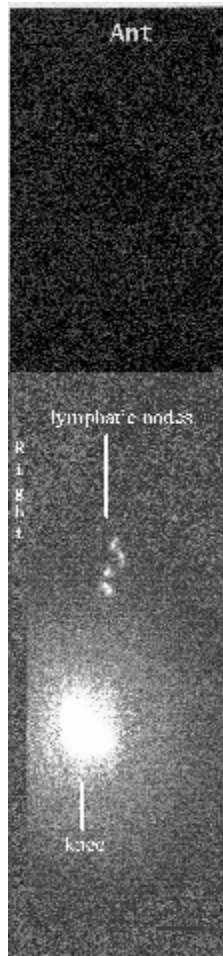
Radionuclide therapy using  $^{153}\text{Sm}$ -EDTMP is recommended as an agent for the treatment of disseminated bone metastases. Clinical results suggest an overall response rate of 78% for this treatment. The administered activity is in the order of 1 GBq, and is limited by the tolerance dose of the red marrow. The rather high yield of the 103 keV gamma rays can be used for retention measurements with the whole-body counter for determining the uptake of  $^{153}\text{Sm}$ -EDTMP in bone.

**Table 1:** Characteristics of the used radionuclides ( $^{166}\text{Ho}$  and  $^{153}\text{Sm}$ )

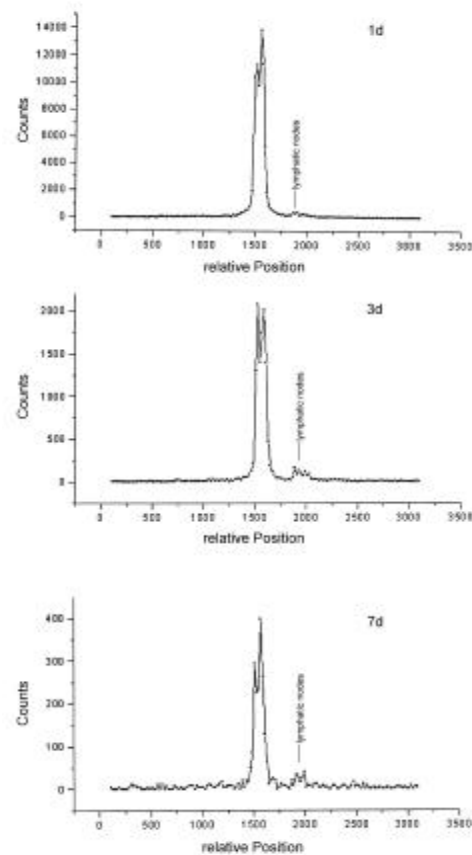
	$^{166}\text{Ho}$	$^{153}\text{Sm}$
Half-life [h]	26.9	46.25
Max. beta energy [MeV]	1.9	0.81
Soft tissue penetration [mm]		
Max.	8.7	2.6
Mean	2.1	0.5
Gamma Energy [keV]	81	103

## Resultsts and Discussion

*<sup>166</sup>Ho-FH treatment:* Patients were scanned 3h and 24h after intra-articular injection of <sup>166</sup>Ho-FH (Fig.1) with a gamma camera and 1d, 3d, and 7d after injection with the clinical whole-body counter in profilescan mode adjusted to the 81 keV gammaline of <sup>166</sup>Ho (Fig. 2). Leakage of <sup>166</sup>Ho could be measured in about twenty percent of the patients. Activity could be localised in lymph nodes which constitutes the only relevant exposure. Leakage was less than 1% of total activity applied. The determination of the energy dose in non-target organs was calculated to less than 1 Gy by the MIRDOSE 3 software [1]. The measurements on the clinical whole-body counter lead us to the assumption, that the leakage of <sup>166</sup>Ho-FH out of the joint is inconsiderable. Further on this scanning device can be properly used for the localization of small areas of activities within the whole body of the patient.



**Fig. 1:** Gamma picture of a patient with a leakage 1 day after injection.



**Fig. 2:** Profilscans of a patient with a leakage 1, 3 and 7 days after injection.

*<sup>153</sup>Sm-EDTMP treatment:* Activity of <sup>153</sup>Sm-EDTMP in bone was determined three weeks after injection with the whole-body counter. Skeletal uptake of this radiopharmaceutical was about 48,2% – 8,1%. The reason for the wide range for the uptake in the skeleton is not sufficiently explainable. A correlation between the percentage of <sup>153</sup>Sm-EDTMP retention and responses to treatment could not be found. However, <sup>153</sup>Sm-EDTMP seems to be a very promising approach for palliative treatment of pain caused by multiple bone metastases which was refractory to common therapeutic strategies.

Measurements show gammalines which do not correspond to <sup>153</sup>Sm but to <sup>152</sup>Eu and <sup>154</sup>Eu. These radionuclide impurities are long-lived (half-life period 13.5 y and 8.6 y) and were produced by neutron activation together with <sup>153</sup>Sm [2]. <sup>152</sup>Eu-EDTMP and <sup>154</sup>Eu-EDTMP show the same metabolism as <sup>153</sup>Sm-EDTMP and it accumulates in bone after each treatment. Activities of both Eu-radionuclides in bone are in the order of 10 kBq after one single treatment, but the radioactive impurities are low enough to account for palliative treatment.

## **Conclusion**

For the determination of the applied dose, the leakage and the quality assurance spectroscopic data of a clinical whole-body counter can be a useful tool for controlling and monitoring in health care.

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# MEASUREMENTS OF IODINE UPTAKE IN THYROID AFTER DIAGNOSTIC ADMINISTRATION OF $^{131}\text{I}$

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## **Introduction**

The administration of therapeutic activity of  $^{131}\text{I}$  radioiodine is usually preceded by a diagnostic administration, done for examination of thyroid function. The quantity of interest is the iodine uptake determined from total activity of iodine in thyroid gland, measured in some time after the ingestion of known activity of  $^{131}\text{I}$ . In some cases, the therapeutic activity, calculated from such data appears to be non-effective, and the patient receives an additional portion of radioiodine. Usually there is no possibility to measure so high activity using the standard clinical equipment and the second dose is given without any measurement of the real content of  $^{131}\text{I}$  in thyroid. It is clear that the increase of the administered activity will cause a great increase of the radiation dose for the patient and also the increasing release of radioiodine from the patient's body. Moreover, the procedure should be considered as unjustified, if the content of the iodine in thyroid already reached its maximum.

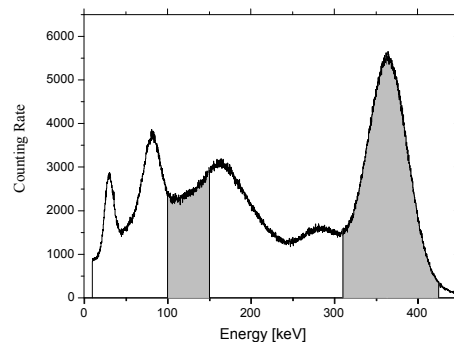
The second administration of  $^{131}\text{I}$  is based on an assumption that the real radioiodine uptake was lower than expected because of one of two possible reasons – either the uptake in the thyroid is lower at high therapeutic doses than at the diagnostic ones, or the value of the uptake was incorrectly determined during the diagnostic procedure.

Sometimes, such incorrect results may appear because the thyroids of the patients are generally far from standard ones, concerning their weight and depth in the neck tissue. The last problem can be at least partly override if the energy spectrum of the radiation emitted from the thyroid is measured. This paper presents the results of calibration of a NaI scintillation counter, with regard to the thyroid depth [1] and the results of measurements performed for a group of patients who received a diagnostic dose of  $^{131}\text{I}$  in Nuclear Medicine Department in Brodno Regional Hospital in Warsaw [2].

## Method

Energy spectrum, of the radiation emitted from the thyroid, includes all the peaks of the  $^{131}\text{I}$  emission spectra, attenuated by the tissue layer between the emission point and the scintillation counter, as well as a broad energy band of Compton scattered radiation. It was expected, that the shape of the spectrum will depend on the effective depth of the thyroid gland, because both attenuation and Compton scattering depend on this depth but in different way.

In order to select the proper parameter of the spectrum, the counting rate was determined for three common peaks of 30, 80 and 364 keV and additionally for the whole spectrum and for the energy band between 100 and 150 keV. Analysis of the results showed that the most accurate way to determine the thyroid depth is to use the value of the ratio of the counting rate for 364 keV peak to the counting rate for the energy band from 100 to 150 keV (see Fig. 1). The plot of this ratio in dependence of the thyroid depth (Fig. 3) has the highest slope, comparing with the other investigated parameters, therefore the value of this ratio is the most sensitive to the change of the thyroid depth. The detector efficiency,  $\epsilon$ , for the 364 keV peak, was determined in dependence on thyroid depth, so the appropriate value can be used for each individual person.



**Fig. 1.** Energy spectrum of  $^{131}\text{I}$  gamma radiation, measured with scintillation detector (NaI). Marked areas under the curve were used for determination of the effective depth of the thyroid gland.

## Calibration of the counter

The thyroid counter from the Institute of Atomic Energy, used for this work, has a NaI(Tl) scintillation detector (Tesla) with beryllium window, mounted in the SS-33W52 counter (manufactured by POLON, Bydgoszcz, Poland).

A special phantom of human neck and thyroid was designed. This is cylinder shaped vessel (128 mm diameter, 165 mm high), with a cover and two small PMMA (13 cm<sup>3</sup>)

cylinder vessels inside the big one (Fig. 2). During the measurements, the big vessel is filled up with distilled water and small ones with solution of iodine  $^{131}\text{I}$  of known activity.



Fig. 2. Water phantom of human neck used in this work.

The energy spectra of the radiation emitted from the phantom, were recorded for 10 simulated depths from 24 to 52 mm, while the distance between detector and the neck phantom surface was always equal to 12 cm. Radiation energy spectrum was measured in the range from 15 keV to 450 keV.

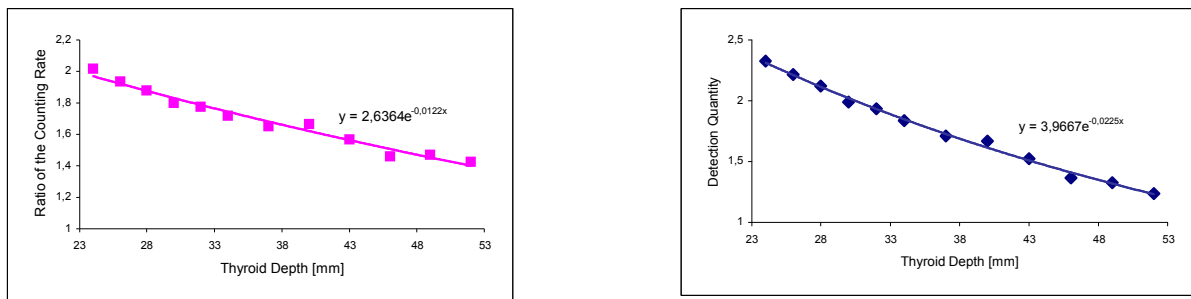


Fig. 3. Left - ratio of the counting rate in 364 keV peak to the counting rate in the Compton scattering band (100– 150 keV), in dependence on the depth of the calibration source in the phantom. Right - ratio of the detection efficiency in dependence on the depth of the calibration source in the phantom.

The resulting dependence of the ratio of the counting rate in 364 keV peak to the counting rate in the Compton scattering band (100– 150 keV),  $S$ , on the thyroid center depth,  $d$ , can be estimated as:

$$S = 2.64 e^{-0.012d} \quad (1)$$

and the efficiency for the 364 keV peak can be expressed as:

$$\varepsilon = 3.97 e^{-0.0225d} \quad (2)$$

## Measurements

During the tests in hospital, the activity of iodine  $^{131}\text{I}$  in thyroid gland was measured about 24 hours after swallowing the pills with isotope. A considerably large number of 98



patients were subjected to the measurements. In all the cases, the routine measurements were performed by the hospital staff and then repeated by the IAE researches using their specially calibrated counter. Time of each IAE measurement was equal to 10 minutes. The counting rate was determined for three common peaks 30, 80 and 364 keV and for some energy bands between them. The ratio of the counting rate in 364 keV to the counting range in energy band from 100 to 150 keV was then used for determination of effective depth of the thyroid gland.

The effective depth was determined using the results of earlier calibration. The ratio  $S$ , of the counting rate in 364 keV peak to the counting rate in the energy band from 100 to 150 keV was related to the effective depth using the equation (1) and the activity of  $^{131}\text{I}$  in thyroid at the time of measurements was calculated from the equation (2).

The iodine activity in thyroid, determined from the spectrometric measurement was always higher than those measured by standard method. In most cases the difference was below 30%, so of no practical importance. However, there was a group of patients (about 20% of the investigated group) who needed more careful consideration. In these cases, the shape of the measured spectra indicated large depth of thyroid, even over 50 mm. In these cases the error of standard measurements of the iodine uptake can exceed 100%.

## Conclusions

The measurements performed up to now, showed that the spectrometric measurements can be useful in selection of the patients who need special consideration during the  $^{131}\text{I}$  diagnostics and treatment. The next step of the work will include the measurements of the real activity of  $^{131}\text{I}$  in thyroid gland, after the therapeutic administration of radioiodine. A special collimator was designed for this purpose and the thyroid counter was calibrated using a phantom with inserts simulating different shapes of pathologically changed thyroid glands. It can be expected that the improvement of accuracy of the diagnostic measurements and better control of real activity of  $^{131}\text{I}$  in thyroid gland after the therapeutic administration will contribute to the process of optimisation of radiation doses to the patients and medical personnel.

## References

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2. J. Osko, T. Pliszczyński "Measurements of iodine uptake in thyroid after diagnostic administration of  $^{131}\text{I}$ " IAE Annual Report 2001, Otwock – Swierk, Poland

# DOSE REDUCTION IN MAMMOGRAPHY AS A CONSEQUENCE OF QUALITY ASSURANCE PROGRAMME

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## Abstract

Mammography became one of more frequent radiological examinations in the most of European countries. This is accepted as a sensitive method for detection of early breast cancers. On the other hand, this is the special imaging technique, where reliability of result is strongly influenced by technical conditions. Hence, the exposure parameters should have been chosen in dependence on image recording system. In opposite, an informative quality of the produced image is not sufficient for a true clinical diagnosis. Thus, optimisation of the practice is not achieved and a benefit for patient is none.

Implementation of Quality Assurance programme can prevent the situation: systematic quality control of mammographic x-ray units allows finding the technical and methodological incorrectness. If these findings are followed by the appropriate remedial actions, the main aim will be achieved, i.e. good quality of image with reasonable low doses to patients.

This approach was verified on 15 mammographic units in Lodz, covered by full cycle of quality control programme. Doses were measured during CC projection for over 100 patients. Quality of image was evaluated for RMI 156 accreditation phantom as a standard object. The main technical incorrectness was found for automatic exposure control system, and especially for compensation of object thickness. The available remedial actions allowed reducing the doses to patients by 21% (on average). Additionally, after the remedial action quality of phantom image was found better for 3 facilities where primary had been below the acceptance level.

The main points of Quality Assurance programme are available for most of x-ray diagnostic facilities and should have been obligatory.

## Introduction

Mammography is one of more frequent x-ray examination in the most of European countries, being used as well for diagnosis of clinical cases, as for screening [1].

Mammography is well known as a sensitive method for detection of early breast cancer, but under condition of high reliability of results; that means an excellent quality of image, which fulfils diagnostic requirements [2]. Real benefit for patient can be achieved when this good quality of image is obtained with reasonably low dose. The same is required by the rules of radiation protection, and means an optimisation of examination.

Implementation of Quality Assurance programme can help to fulfil the above requirements: systematic quality control of mammographic x-ray units allows finding the technical and methodological incorrectness. If these findings are followed by the appropriate remedial actions, the main aim will be achieved, i.e. good quality of image with reasonable low doses to patients.

This approach was verified on 15 mammographic units in Lodz, covered by full cycle of quality control programme. The paper presents methods and results of the control; the most frequent incorrectness and possibility of their repairing are also discussed.

## Methods

Quality Assurance routine actions include:

- 1) a control of entrance surface air kerma to patients,
- 2) a control of images of accreditation phantom (together with rejection analysis of clinical mammograms),
- 3) Quality Control (QC) tests,
- 4) the proper remedial actions (undertaken if necessary),
- 5) repeated control of doses, images and QC test for checking an effectiveness of corrections.

Despite of rejection analysis, all these actions were involved into the study discussed here.

Entrance surface air kerma (ESAK) was measured directly on breast of patients. Thermoluminescent dosimeters (TLD) made in Poland from LiF, being analogue of TLD-100 were used for that purpose. Set of 5 TLDs packed into thin plastic sheet was placed directly on breast during CC projection in standard localization. TLDs were read using Laboratory Reader-Analyser TL RA94 (Mikrolab, Poland). The dose response as well as the energy characteristics of the TLD's had been evaluated before the study.

To avoid variation of doses resulting from natural differences in breast thickness, the measurements were performed only for patients whose thickness of compressed breast was ranged from 4.5 cm to 5.5 cm.

The kVp, mAs and thickness of compressed breast were recorded for each patient.

Entrance surface air kerma (ESAK) were estimated together with an error ( $\Delta$ ESAK).

$$ESAK = a \overline{N.C.}$$

$$\Delta E_{SAK} = \frac{\overline{N.C.}}{100} \sqrt{[C.V.(N.C.)]^2 + [C.V.(a)]^2}$$

where:

a - dose - response coefficient,

N.C . - mean value of the number of counts,

C.V.(N.C.) - coefficient of variation of N.C.,

C.V. (a) - coefficient of variation of a.

Quality of image was controlled using accreditation phantom RMI 156. This is designed to test the performance of a mammographic system by a qualitative evaluation of the system's ability to image small structures similar to those found clinically and being important in the early detection of breast cancer. The phantom includes 16 various sets of test objects: six different size nylon fibers simulate fibrous structures in ducts (from 0.4 mm to 1.56 mm), five groups of simulated micro-calcifications (from 0.16 mm to 0.54 mm) , and five different size tumor-like masses (from 0.25 mm to 2.0

mm). All the objects are inserted in wax block. The phantom is exposed like as a breast in CC projection. Phantom images were analysed by physicists.

Each test object in the phantom image is given a score of one. If only a portion of an object or fraction of the simulated structure is seen, half scores may be used. The criteria given by the phantom producer (based on the ACR criteria) requires a score of 4 for fibrils, 3 for simulated micro-calcifications and 3 for simulated tumour masses. Consequently, the minimum for acceptance of image quality is the score 10.

The parameters being controlled in frame of Quality Control tests are listed in Table 1, where the proper action limits in accordance to European guidelines [3] are also given.

**Table 1. Parameters (or functions) of mammo-units involved in QC tests**

<b>Parameter/function</b>	<b>Action Limit</b>
<b>kVp accuracy</b>	<b>Differences &gt; 1 kV</b>
<b>kVp reproducibility</b>	<b>Coefficient of variation &gt; 0.02</b>
<b>HVL (28 kVp)</b>	<b>HVL &lt; 0.31 mm Al or HVL &gt; 0.40 mm Al</b>
<b>AEC: compensation of kVp</b>	<b>Maximal difference &gt; 0.1 mean O.D.</b>
<b>AEC: compensation of thickness</b>	<b>Maximal difference &gt; 0.1 mean O.D.</b>
<b>AEC reproducibility</b>	<b>Coefficient of variation &gt; 0.05</b>
<b>ESAK reproducibility</b>	<b>Coefficient of variation &gt; 0.05</b>
<b>Film/screen contact</b>	<b>Visible areas of poor contact on the film</b>
<b>Uniformity of compression force</b>	<b>Difference of distances &gt; 1% SID</b>
<b>Luminance of viewing boxes</b>	<b>Luminance &lt; 2000 cd/m<sup>2</sup></b>

## Results

Fifteen mammo-units in Lodz were included into the study, and submitted to full cycle of quality control programme. The first stage of the cycle was followed by proper remedial actions, and then all measurements were repeated.

Values of doses measured in the both stages are shown in Table 2.

Table 2. Doses to patients in mammo-units before and after remedial actions.

	Before	After
Number of patients covered by measurements	135	142
Range of dose values	4.4 mGy – 22.7 mGy	4.5 mGy – 18.4 mGy
Geometrical mean	8.9 mGy	7.5 mGy

Results of QC tests are presented in Table 3.

Table 3. Summary of results of QC tests for controlled mammo-units.

Parameter/function	Number of units in acceptance range	
	before	after
Voltage-accuracy	5	13
Voltage-reproducibility	15	14
HVL	13	13
AEC-compensation of kVp	2	9
AEC-compensation of thickness	1	2
AEC-reproducibility	15	14
Uniformity of compression force	11	14
Film/screen contact	8	14
Acceptability of image of accreditation phantom	9	14

After remedial actions only 14 mammo-units was evaluated, as one was excluded from usage because technical state found during the first QC tests.

Better results of the second QC tests were possible as an effect of remedial actions, which are listed in Table 4; a reduction of doses achieved for the particular mammo-units is also shown here.

**Table 4. Remedial actions and dose reduction for the controlled mammo-units.**

<b>Mammo-unit</b>	<b>Remedial actions</b>	<b>Dose reduction</b>
Al.-1	AEC-adjustment, selection of cassettes	7.4%
Al.-2	Generator regulation, AEC-adjustment, selection of cassettes	5.9%
Al.-3	AEC-adjustment, compression plate	18.5%
Be-1	Generator regulation	7.05
Be-2	Generator regulation, AEC-adjustment, selection of cassettes	21.4%
Be-3	Generator regulation, AEC-adjustment, selection of cassettes	17.4%
Ph-1	Generator regulation, AEC-adjustment, selection of cassettes, compression plate	15.6%
Ph-2	AEC-adjustment, selection of cassettes	18.3%
Ph-3	Generator regulation	18.9%
Ph-4	Compression plate	13.0%
El-1	AEC-adjustment	1.8%
El-2	Generator regulation, AEC-adjustment, selection of cassettes	11.7%
So-1	Generator regulation, AEC-adjustment, selection of cassettes	17.9%
So-2	Generator regulation, AEC-adjustment, selection of cassettes	1.9%

## Conclusions

### At the first stage of control:

doses to patients differ more than 4 times, despite of similar breast thickness and are close or higher than reference level for CC projection (7 mGy) for most of controlled mammo-units, even one from controlled units passed all QC tests successfully before technical corrections; non-acceptable quality of image of accreditation phantom was found for 6 (from 15) mammo-units.

### At the second stage of control:

quality of image of accreditation phantom was better and acceptable for all of the mammo-units (except of one excluded from usage),  
geometrical mean of patient doses decreased 16%.

### General remark:

Real benefit for patients in mammography is possible only when main points of Quality Assurance programme are obligatorily fulfilled in mammo-units.

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# QUALITY CONTROL OF CONVENTIONAL DIAGNOSTIC RADIOLOGY EQUIPMENT IN SERBIA AND MONTENEGRO

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## Introduction

Contribution of radiology to better diagnoses and treatments is evident. In parallel, efforts were oriented towards the improvement and control of equipment. The importance of quality assurance (QA) of diagnostic X-ray equipment is well recognized. Application of QA programme is very important when optimization of image quality and reduction of patient exposure is desired. In Serbia and Montenegro a substantial effort to establish national standards and basic quality requirements has been done, unfortunately, with partial results. In diagnostic radiology, for the success of the programme, it is essential to apply scientific and technical knowledge to definition of clear protocol for the safety and measurements of the performance of X-ray units.

There are more than 1500 diagnostic X-ray tubes in service in Serbia and Montenegro. Diagnostic X-ray equipment is checked on annual basis. X-ray equipment QC protocols have been adopted from several international standards and guidelines [1,2], which have been modified according our practice and measuring equipment. The survey protocol is mainly oriented to the control of technical parameters of radiological equipment, and efforts to include patient dosimetry and image quality have are in progress. Some findings from checks on X-ray units are presented below. According our inventory, about one half of all installed units is used for conventional X-ray diagnostics, 10% are mobile units, 2% of all in units in operation are mammographic devices an another 2% are computed tomography units. In addition there are about 600 dental X-ray units, which is 36% of all units. It is worth mentioning that almost 30 % of all installed units have single-phase generators, another 25% are tree phase (six and twelve pulse) generators and nearly 9% are high frequency units. Majority of units was installed more than 25 years ago [3]. Also, the major problem is still existing fluoroscopy with fluorescent screens in the place of image intensifier.

## Materials and methods

The Quality Control (QC) in radiography is a central part of QA programme, which deals with equipment maintenance and monitoring. QA in diagnostic radiology is a mean of maintaining standards in imaging and working towards minimizing patient and staff doses. To accomplish these objectives, a number of physical parameters that affect the performance of X-ray imaging system are to be measured. Some of them, followed by basic criteria are shown in Table 1. There are four stages to the checks and measurements applicable to X-ray imaging equipment: Critical examinations, Acceptance, Commissioning and Routine performance testing [2]. As a part of QA program in diagnostic radiology, the performance characteristics of 96 conventional X-ray units were measured in six months period during 2003. The diagnostic units were located in hospitals all over the country. They represent 25% of the total conventional diagnostic units of the country. About one half of measured X-ray units were new-installed.

Table 1. Acceptability criteria for QC in diagnostic radiology [1,2]

Parameter	Good	Normal	Poor
Tube voltage accuracy (kV or %) against dial figures and variation	< 5 kV	( 5 kV, 10 kV)	$\geq$ 10 kV
Exposure time accuracy (t>0.1 s)	< 5 %	( 5 %, 15 %)	> 15%
Output (O/P) at 80 kV at 1 m ( $\mu$ Gy/mAs):	43-52	26-43, 52-69	<26, >69
reproducibility	< 15%	( 15 %, 25 %)	> 25%
variation with kVp (n) O/P (kVp) <sup>n</sup> where n:	2.0-2.2	(1.7-2.2, 2.2-2.5)	<1.7, >2.5
linearity mA	5 %	10 %	non-linear
Filtration mm Al	2.5-3.5	-	<2.5, >3.5
X-ray/light beam alignment	< 1 %FFD	( 1%FFD, 2%FFD)	$\geq$ 2% FFD
Image intensifier spatial resolution (lp/mm)	$\geq$ 1	-	<1
Dose rate in fluoroscopy (mGy/min)	<25	(25, 50)	$\geq$ 50

The peak voltage was measured for nominal values from 50 to 100 kV (in steps of 10 kV), using Kietley kVp Meter Model[4]. Differences between displayed and actual values are recorded. The total filtration (inherent and additional) was derived from the determination of the half value layer (HVL) of the actual beam at 80 kV, according to the procedure recommended elsewhere [1]. The measurements of tube output were performed at 50-100 kV (in 10 kV steps) for 100 mAs at 1m focus-to-detector distance, using Victoreen Rad Check Plus Model 526 ionizing chamber. The measurement at 80 kV was repeated for at least 5 times, to determine reproducibility of the tube output. The linearity of the output (variation with mA) was checked between 10 mA and 200 mAs. The exposure time reproducibility and accuracy for five time settings (from 0.1s to 1s) were checked using Kietley timer Model. The correspondence between the light beam and the actual X-ray beam was measured using appropriate test plate by Wellhofer [6]. All fluoroscopic units tested, were equipped with image intensifier and Automatic Exposure Control (AEC). Entrance surface air kerma rate was measured using Victoreen Rad Check Plus ionizing chamber and standard dosimetric water phantom [1]. Spatial resolution was evaluated using test plate by Wellhofer. All measuring equipment was calibrated in traceable Secondary Standard Dosimetric Laboratory in VINCA Institute of Nuclear Sciences.

Patient doses were calculated from the X-ray tube output, using radiographical technique data for “standard” patient (70 kg) in each hospital in question. This indicative dose assessment for most frequent radiographic examination was used to compare reference dose values. The dosimetry method involves a measurement of X-ray tube output (e.g. air kerma at defined geometry for a range of tube voltages) followed by the use of backscatter factor (BSF) data and geometry corrections to determine the entrance surface dose. This methodology enables a relatively large number of patient dose estimates from a small number of measured parameters, the measurements are part of a QC program.

## Results

The results are presented in Table 2. For all units, tube output dependence on  $kV^2$  was linear with  $r = 0.99$ . It is important to notice that about one half of the measured X-ray units are new installed and results for these units are results of acceptance testing. It is presented in the Table 2 that 96 % of all imaging facilities could image 1 lp/mm or better, which is considered to be acceptable for standard television systems. Spatial resolution for new equipment was

better than 2 lp/mm in all facilities. Same conclusion is valid for entrance surface dose rate in fluoroscopy which was (10.4 ± 0.5) mGy/min for all new-installed pieces of equipment.

Table 2. Results of QC for 95 conventional X-ray units

Parameter	Acceptability criteria		
	Good	Normal	Poor
Tube voltage accuracy	64%	23%	13%
Timer accuracy	63%	24%	13%
Output at 80 kV at 1 m ( $\mu$ Gy/mAs)	5%	21%	71%
reproducibility	98%	1%	1%
linearity	62%	22%	16%
Filtration mm Al	30%	-	70%
X-ray/light beam alignment	68%	23%	9%
Image intensifier spatial resolution	96%	-	4%
Dose rate in fluoroscopy	94%	4%	2%

The confidence limits of the exposure values calculated from the collected data are presented in the Table 3. Mean, standard deviation values and coefficient of variation were obtained. The mean consistency results for some parameters (with standard deviation and coefficient of variation) are presented in Table 3. As, the new installed units are high frequency, extracting related results from the bulk, it can be shown that they are significantly more consistent than older X-ray units. Also, one of the major shortcomings of older X-ray units was inadequate beam filtration, which was not the case with new equipment

Table 3. Consistency of five measured parameters in routine and acceptance testing

Parameter	Acceptance testing			Routine testing		
	Mean	Standard deviation	Coefficient of variation	Mean	Standard deviation	Coefficient of variation
Tube voltage accuracy (%)	1.4	0.8	0.5	8.6	6.8	0.8
Timer accuracy (%)	3.2	0.9	2.2	11.5	13.2	11.1
O/P at 80 kV at 1 m ( $\mu$ Gy/mAs)	85.8	13.3	0.16	73.4	35.6	0.5
reproducibility (%)	0.3	0.7	2.1	1.6	1.9	1.2
linearity (%)	1.7	1.1	0.7	15.5	18.7	1.2

A great variation in patient doses were found in patient dose survey (Table 4). Some reasons for the variations became apparent, as speed class of film-screen combination, which was 200-400, and manual exposure control settings. The typical technical factors used vary by a wide range. For instance, X-ray tube voltage extends from 60 to 117 kVp, for chest radiography and there is a tendency of smaller product of tube current and exposure time for high tube voltage, which provides lower entrance surface dose. Also, other equipment related, technologically limited, factors also affect patient dose in all examinations. These are three phase generators, insufficient beam filtration and manual exposure control setting. Distributions observed for various dose quantities are typically skewed, with mean values generally greater than corresponding medians, so small number of patients receives high doses. Except for chest and skull PA examination, all estimated doses are less than stated reference levels. The explanation for relatively high doses lies down in comparison of actual practice with example of good radiographic technique [6].

Table 4. Means(with range in parentheses) of patient weight and applied tube voltage, means (with range in parentheses), medians and third quartile values of entrance surface doses and mean effective doses for different radiographic examinations.

Examination	Tube voltage (kVp)	Entrance surface dose (mGy)			Diagnostic Reference Level [4] (mGy)
		Mean	Median	Third Quartile	
Cervical spine AP	65 (45-90)	2.3 (0.4-6.0)	2.2	2.7	*
Cervical spine LAT	65 (45-90)	2.1 (0.4-5.2)	1.9	3.0	*
Pelvis AP	69 (55-85)	4.7 (1.2-18.5)	3.0	4.4	10
Thoracic spine AP	74 (55-90)	5.2 (1.1-11.7)	4.1	6.9	7
Thoracic spine LAT	81 (55-110)	9.0 (1.5-34.2)	5.8	11.0	20
Lumbal spine AP	76 (63-90)	8.2 (1.7-29.9)	4.8	11.8	10
Lumbal spine LAT	88 (70-125)	16.5(1.9-41.0)	8.2	30.0	30
Chest PA	73 (52-95)	0.4 (0.1-0.9)	0.3	0.7	0.3
Chest LAT	87 (60-117)	1.0 (0.3-2.4)	1.0	1.3	10
Skull PA	68 (50-85)	3.8 (0.6-12.5)	2.5	5.2	5
Skull LAT	66 (45-96)	4.1 (1.0-13.4)	2.3	4.2	3

## Conclusions

Although the use of ionizing radiation in medicine is controlled by several pieces of legislation, the legal framework does not cover the area of QA and QC. Concerning that fact, in addition to legal documents, several international guidelines are used. In spite of this, there are still many unsatisfactory aspects in our practice. It is therefore, necessary to include patient dosimetry and image quality in the quality management system, which should be in operation in every diagnostic radiology department. In our experience, QC program has positive effect on X-ray equipment performance in a period of a few years. It is essential to perform QC tests for all installed X-ray units at least on annual basis. This would lead to production of consistent X-ray images, with minimal retake rate and hence, will contribute to decreasing of the patient dose. Patient doses are determined by multitude factors which interact in very complicated manner. It is very important to perform real patient dose measurements in hospitals. These results are link between patient dosimetry, as a first step in optimization of radiation protection, and quality assurance program in diagnostic radiology. The use of dose assessment from X-ray tube output for standard patient enables comparison of dosimetric data when it is not possible to perform extensive measurements using hundreds of patients. It is of great importance to extend the survey to large number of hospitals and to include complex examinations, with goal to establish diagnostic reference levels on national scale.

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# QUALITY ASSURANCE IN DIAGNOSTIC RADIOLOGY IN HUNGARY – FIRST EXPERIENCES IN ACCEPTANCE TESTING

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## **Introduction: The importance of technical quality assurance**

It is a general experience that optimum imaging with minimum patient doses, moreover, the safe operation and long life of X-ray equipment can be assured by regular measurement of technical parameters and checking of their constancy (routine performance testing) only. These tests are generally known as *quality control*, while together with the so-called corrective actions and its management it is called (physical-technical) *quality assurance* (QA).

## **Regulation**

In the European Union, Directive 97/43/EURATOM about radiation protection of patients requires – among others – the good practice of (physical-technical) quality assurance. In Hungary, Decree No. 31/2001. (X.3.) of the Minister of Health harmonizes all of its requirements. *Acceptance testing* of new diagnostic X-ray equipment is assigned to NPHC-NRIRR. Further full performance testing (*status testing*) is required yearly and after major servicing, it will be the task of accredited testing organizations. Simple daily checks (the so-called *constancy testing*) are the responsibility of users (licensees) themselves (see Table 1). QA programmes are under the surveillance of the radiation health authority (radiation health departments of State Public Health and Medical Officer Service).

## Preliminaries

QA has been a daily practice in radiation therapy and nuclear medicine for a long time. A National Patient Dose Assessment Programme has also successfully run since 1989. We had, however, only few preliminaries in QA in diagnostic radiology in the second half of the eighties. Nowadays there are running QA programmes in some hospitals and mammography centres.

## Start and benefits of acceptance testing

According to generally accepted definitions, the main aim of acceptance testing is checking the compliance of the equipment with the contract and/or manufacturer's specifications (and local regulations). It is the first benefit of the user. In the – unlikely but in principle possible – case of non-compliance the user can successfully complain using the test report. The testing activity of our institute is independent from manufacturers, it is run within the frame of an accredited testing laboratory, using calibrated measuring instruments and based on valid international standards. The other benefit for the user is the measuring of the so-called base levels for further QA, i.e. initializing of a QA programme assuring long, safe and optimum performance of equipment.

*Table 1: QUALITY CONTROL AND SAFETY TESTS IN DIAGNOSTIC RADIOLOGY*

Test	Regulation	It is obligatory for	Its characteristics, required frequency	It is made by
Acceptance test	Decree No. 31/2001. of the Minister of Health, 12.§ (2)	New X-ray equipment	QA: complete status evaluation, Before putting into use (single test)	NPHC-NRIRR
Status test	Decree No. 31/2001. of the Minister of Health, 13.§ (2)	X-ray equipment after acceptance testing or QA programme started	QA: complete status evaluation, Yearly and after greater maintenance	Bodies to be accredited
Constancy test	Decree No. 31/2001. of the Minister of Health, 13.§ (2)	X-ray equipment after acceptance testing or QA programme started	QA: routine check, Daily, weekly, monthly etc. (under elaboration)	Licensees (users)
Periodic safety check	Decree No. 47/1999. of the Minister of Health, 17.§ and Annex 13.	All functioning X-ray equipment, except CT and dental equipment	Electrical, mechanical and radiation safety, Interventional X-ray: yearly, other: bi-yearly, and after greater maintenance	Notified bodies and bodies authorized by decision of Authority of Medical Devices
Radiation protection test (till full EU-membership only)	Decree No. 16/2000. of the Minister of Health, 4-5.§ and Annex 3.	Every X-ray equipment type	Single (type) test, Before the first installation	NPHC-NRIRR

### **Status testing and constancy testing**

A *status test* means a full performance measurement equivalent to an acceptance test although some simplifications are possible. (E.g. checking of documents and measurement of leakage radiation or tabletop attenuation can be omitted.) We hope that first accredited bodies will start their work in the next few months. In an intermediate phase status testing is likely to be performed mostly by service firms as nowadays only they possess the required test devices. The long-term aim is to organize – and provide with test devices – an independent testing network, based on medical physicists, similarly to Great Britain or Sweden. We are of the opinion that – at least in the first some years – the extent of *constancy testing* has to be restricted to the simplest checks, both for financial and professional skill reasons. The elaboration of QA programmes for diagnostic radiology is in progress, with the co-operation of NPHC-NRIRR and National Board of Radiology. As total replacement of the equipment park is likely to need more than 15 years, initializing QA programmes on existing equipment is also to be solved.

### **Periodic safety checks**

A special requirement in Hungary – originating not from the EU but similar to the system in Germany – is that medical devices are to be checked periodically from the point of view of electrical, mechanical and radiation safety. For X-ray equipment it is required biannually and after major servicing, except in case of interventional equipment for which it is required yearly. This checking is performed by organizations authorized by the Authority of Medical Devices and under its surveillance. After overcoming some initial difficulties this system has been functioning successfully for three years. This check is independent from QA testing although they are overlapping to some extent.

### **Radiation protection testing of equipment**

It is a type test which is, by regulation, a necessary condition for selling an equipment type in Hungary. After the full EU-membership of Hungary it will be stopped as CE-marking certifies also radiation protection aspects of conformance to Annex I of EU Medical Devices Directive. Table 1 outlines all QA and safety tests in diagnostic radiology.

### **Practice and results of acceptance testing**

Acceptance testing is based on EN/IEC 61223-3 standard series. The needed universal X-ray parameter measuring devices and other test devices were bought with the financial support of the EU. Acceptance testing has been continuous since May 2002. During the first year about 35 acceptance tests were performed. From them 2 were mammography equipment,

the others are radiographic and fluoroscopic equipment, including tomographic and mobile equipment and surgical image intensifiers. About in every third case some need for servicing or adjustment was detected. These were recorded not only in the test reports but also in official letters written to the competent leaders of the hospitals. As it always happens in the warranty period, these servicing or adjustments are free of charge for the hospital. We mention that with our measuring instruments we could detect differences between specified and actual filtration or between displayed and actual X-ray tube current. Tests are always non-invasive.

### **Objective difficulties**

Modern X-ray equipment have the general characteristics that the possibility of operator errors are minimized by their construction. Taking into account the non-invasive character of the tests, it is not possible to go into a so-called “service mode” if no authorized service personnel is present. Some examples for the difficulties arisen from it: we have to remove and take back cassette or enter “patient data” before every exposure. In some cases there was impossible to measure tabletop attenuation as without the presence of a radiographic cassette exposition is not possible. Moreover, many times only the current time product can be selected, the tube current is not independent.

### **Subjective difficulties**

As everything at the beginning, acceptance testing is not known very much by radiological personnel although in every case we send an information leaflet in advance. In most cases the cost of the tests is the main problem for hospitals. Although we insist on the presence of the representative of the customer – mainly from the point of view of avoiding later legal disputes – personnel often say that we can perform measurements without them.

Radiographers often did not know how to use fluoroscopic or tomographic equipment so we had to rely on their documents. In general, knowledge of radiographers is very different; in many cases they do not know even the type and speed of the intensifying screens used by themselves. So improving the education and training of radiographers as well as training them on QA is an absolutely urgent and necessary task. Further, mostly organizational problems occurred in testing surgical image intensifiers in operating theatres.

### **Conclusions**

So the started way of implementing QA in diagnostic radiology needs a lot of further efforts, adapting experiences of other countries, and also some financial help to reach an acceptable level in the EU.



# OPTIMIZATION OF RADIATION PROTECTION AT BOHUNICE NPP

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## Introduction

Bohunice Nuclear Power Plant is situated in south - western part of Slovakia about 50 km away from Bratislava. There are four PWR reactors 440 MWe each - two units with reactors VVER - 230 (V1 NPP) and two units with VVER - 213 (V2 NPP).

The latest legislative changes in the optimization of radiation protection in Slovak republic had been done at the end of 2000. The law No 470/2000 novelized the state law No 272/1994 „Protection of People Health“ and subsequent Code No 12 was issued. New legislative introduced new terms as dose constraints and financial equivalent of man Sv

## New legislative requirements

The Law No 470 defines the optimization as: “Each person who performs the activity that leads to the exposure is obliged to ensure the magnitudes and likelihood of exposures and the numbers of individuals exposed be as low as reasonably achievable, economic and social factors being taken into account”.

All other requirements for the optimization process are given in the mentioned Code No12 of Ministry of Health.

Code 12 stipulates the technical and organizational requirements for proving the Rational Achievable Level (RAL) of radiation protection. This level can be proved by means of the comparison of the dose distribution to the costs of protection. The Code also defines the dose constraints that are used in the process of proving the RAL. An example of two figures of dose constraints is:

- collective dose 20 man mSv for the specific task,
- individual exposure 1 mSv per day

The values of the financial equivalents of personal exposure – so called the alpha coefficients - are used for the calculation of the benefit of proposed measures. Alpha coefficients and individual exposure ranges are presented in following table.

Alpha coefficients	Individual exposure
2 mil. Sk / man Sv	less 2 mSv/y
5 mil. Sk / man Sv	2 – 5 mSv/y
15 mil. Sk / man Sv	5 – 15 mSv/y
20 mil. Sk / man Sv	15 – 30 mSv/y
25 mil. Sk / man Sv	30 – 50 mSv/y

Comparison of alpha coefficients expressed in US\$:

USA (NRC)	Switzerland	Czech republic	Slovak republic
200 000	3 000 000	17 000 – 170 000	50 000 – 750 000

### Impact of legislative changes into Bohunice NPP

Changes in the legislative of Slovak republic influenced the radiation protection system in Bohunice NPP.

As a first step the changes were implemented into the **plant's internal documentation**.

At the same time **new software modules** were creating in order to clear recording, evaluation and finally to have a tool that supports the discussions with the regulatory body. The software modules have been a part of Integration Information System of Bohunice NPP. The IIS software is the same for all Slovak NPPs.

**Revision of plant documentation** consisted of revision of existed quality assurance system and technological procedures.

New terms were defined, as the Code 12 required it. The dose constraints and internal ALARA criteria were specified and optimization system was also revised. The optimization criteria were the individual exposure per task and per day.

As far as the revision of technological procedures is concerned, all the procedures were supplemented with the dose assessment and analyzed by the ALARA specialists and finally approved by the manager of radiation protection department.

Important role in the proving the Rational Achievable Level of radiation protection plays **the software modules of Radiation Protection** within the Integration Information System of Slovak Electricity. One of the modules is directly devoted to the optimization. Its goal is transparent recording of optimization and to help the evaluation of optimization from Radiation Work Permits after performing the works. ALARA module is linked with other modules as "Operative Dosimetry", "Radiation Work Permits", "External Dosimetry", "Internal Dosimetry",

ALARA software module was put in operation in 2001

## Optimization process

The optimization process at Bohunice NPP can be divided into 2 groups – activities that require approval of ALARA committee and those that can be performed without it. In both cases all facts must be recorded.

If the optimization criteria are exceeded, ALARA committee analyses the work, accepts the adequate measures and sets the task specific dose constraint. The activity is carefully followed during the work on Radiation Work Permit and if any anomaly is revealed, corrective measures are accepted.

After finishing the work the real exposures are computed. If the real exposure is higher than approved by ALARA committee, the “post ALARA process” starts to find the reason and learn from that.

All the plant activities must be optimized. The activities could be divided into standard, repeated works and new, planned works as modifications and special programs. All standard and repeated activities were optimized for the first time.

All facts of optimization process are recorded into the database, which is a part of the software module ALARA. OPTI RP software obtained from CEPN is used for quantitative analysis of optimization. The sensitivity analysis is an another important tool used in optimization process.

The picture below shows an example of the screen used by ALARA specialist. The top line identifies the case. The upper window contains the calculated and by the ALARA committee approved task specific dose constraints and the lower window provides the real exposures:

The screenshot shows the ALARA software interface. The title bar reads 'Arsoz VP-Produktívny mandant EBO : ALARA - [VYHODNOTENIE ŠPECIALISTOM ALARA]'. The menu bar includes 'Vykonaj', 'Edituj', 'Spracovanie', 'Prehľad', 'Blok', 'Záznam', 'Položka', 'Vyhľadaj', 'Okno', and 'Pomoc'. The toolbar contains various icons for file operations and navigation. The main window has tabs for 'ALARA', 'Doplnok', 'Dávky', 'Zhodnotenie po', and 'Podmienky prác'. The 'Dávky' tab is active, showing two data tables.

**ALARA**  
 ALARA: 19 Rok: 2002 Stav: 80 Evidenčné číslo PZ: PZ 2248/01/1

**Dávky - Optimalizovaný odhad.**

Parameter	Jed.param.	Názov parametra	Hodnota
IED	mSv	Priemerná individuálna efektívna	2,13
KED	mSv	Kolektívna efektívna dávka	630,67
MAXIED	mSv	Maximálna individuálna efektívna	10
MAXIEJD	mSv	Maximálna individuálna jednoráz	,96
POCET OSOB	osob	Pocet osôb.	295

**Dávky - Skutočnosť.**

Parameter	Jed.param.	Názov parametra	Hodnota
IED	mSv	Priemerná individuálna efektívna	,93
KED	mSv	Kolektívna efektívna dávka	62,46
MAXIED	mSv	Maximálna individuálna efektívna	3,69
MAXIEJD	mSv	Maximálna individuálna jednoráz	,78
Pocet osôb	osôb	Pocet osôb	67

Vypočítaj ...

Next list contains all the items that enter into the calculation of the protection costs:

- shielding
- decontamination
- waste processing
- protective aids
- salaries
- $\alpha$  coefficients

First five lines are the costs of protection. The last line is the alpha coefficients used for evaluation of financial equivalent of person exposure

Presentation will provide the examples of real optimization cases done at Bohunice NPP. The typical ones are works in steamgenerators (modification of upper feedwater distribution pipelines)

The table below contains the calculated and real exposures for the activity “Exchange of upper feedwater distribution system inside steam-generators at V1.

	Dose assessment before optimization	Dose constrains for SG16 after optim.	Real doses at SG16	Dose costrains for SG26	Real doses at SG26
<b>Collective exposure [man mSv]</b>	<b>101,52</b>	<b>57,25 86,75*</b>	<b>25,18 39,71*</b>	<b>40,55 55,11*</b>	<b>28,60 35,57*</b>
<b>Max. Individual exposure [mSv]</b>	<b>8,46</b>	<b>4,77</b>	<b>3,92</b>	<b>4,06</b>	<b>3,97</b>
<b>Max. daily exposure [mSv]</b>	<b>0,564</b>	<b>1,04</b>	<b>0,92</b>	<b>1,00</b>	<b>0,99</b>

## Conclusion

Apparently the new law and the associate code created a base of transparent and understandable policy of radiation protection and optimization in Slovak republic. The radiation protection legislative was implemented into the praxis and persons became familiar with it. Defining clear and unambiguous terms facilitated the communication between users and the regulatory body – State Health Institute.

Optimization was generally accepted by the workers and managers and began to be a part of safety culture of operation at nuclear power plants.

# QUALITY ASSURANCE/QUALITY CONTROL FOR TESTING OF RADIOACTIVITY

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## **Introduction**

The Testing laboratory of material radioactivity was founded in 1999 on the basis of former Nuclear Research Centre. The primary aim was to test the waste, radioactive polluted materials and apparatus from the shut down nuclear reactor. In 2000 National Accreditation Service of Latvia (LATAK) accredited laboratory. In the following years the enlargement of testing sphere occurred.

The aim of this work is the introduction of quality assurance and quality control according to ISO/IEC 17025 standard in the Testing laboratory of material radioactivity State Enterprise "RAPA" in Salaspils, Latvia.

Radioactivity of Cs-137, Co-60, U-238 and Th-232 trees, as well as other radionuclides is measured using the high performance gamma-spectrometry. Testing of different environmental samples and building materials as well as samples irradiated with neutron flux of nuclear reactor and radioactive polluted objects is carried out. Alfa- and beta-radioactivity of waters is tested using OXFORD Tennelec Solo counting system. Other equipment is used to test the radioactive contaminated surfaces, the testing of tritium activity is done with LSC. The determination of dose rates in workplaces as well as the checking of individual protection for X-ray in medicine are being performed too. The National Accreditation Service of Latvia (LATAK) has accredited all these test methods. The performed measurements are traceable to national and international standards.

## **Results and discussion**

Starting with 2000, laboratory works at the extension of testy sphere as well as on the introduction of the quality system according to the ISO/IEC 17025 standard. During this time 5 new methods were accredited, the Quality Manual was revised, the range of new procedures, instructions was worked out, the documentation follow-up and control was arranged, the number of calibrated equipment has increased, the environmental conditions are strictly observed in the laboratory. The laboratory personal raises their qualification taking part in seminars organized by the IAEA and conferences organized by the EUROLAB devoted to quality control and

assurance, determination of uncertainties, et. al. as well as in events organized by the national institutions.

The credibility of obtained results is ensured by the quality control and quality assurance. The main requirements involved in the quality control and quality assurance of the laboratory according to the ISO/IEC 17025 standard are the following:

- the use only of calibrated equipment;
- regular and long-time use of the reference materials in the control of equipments;
- regular participation in the intercomparison projects;
- internal audits;
- the control charts of background;
- estimation of uncertainty sources;
- calculation of uncertainties within the given interval of credibility;
- document control;
- development of procedures.

Since 1999 we regularly take part in the international and national intercomparison projects (tab.1) results of which are represented in /1,2,3/.

Table 1

Participation of laboratory in the intercomparison projects

<b>Year</b>	<b>Organizer</b>	<b>Object</b>	<b>Radionuclides</b>
1999	Risoe National laboratory, Denmark	Sediment, milk, meet, seaweed, hay	K-40, Mn-54, Co-60, Cs-137, Ra-226, Th-232
1999	National Metrological Center of Latvia	Building materials	K-40, Ra-226, Th-232
2000	Risoe National laboratory, Denmark	Dry milk, aerosols, soil, seaweed	K-40, Cs-137, Co-60, Mn-54, Ra-226, Th-232
2000	National Metrological Center of Latvia	Ion resin, soil, soot	Eu-152
2002	IAEA, Vienna	Mineral matrix	Mn-54, Co-57, Co-60, Cs-134, Cs-137, Eu-152
2003	Risoe National laboratory, Denmark	Sea water, lake water, milk, mineral matrix, sea weed	Cs-137, K-40

As example, the Fig. 1 has demonstrated the comparison of the laboratory's reported values for seaweed with the statistically weighted means of all participants accepted results.

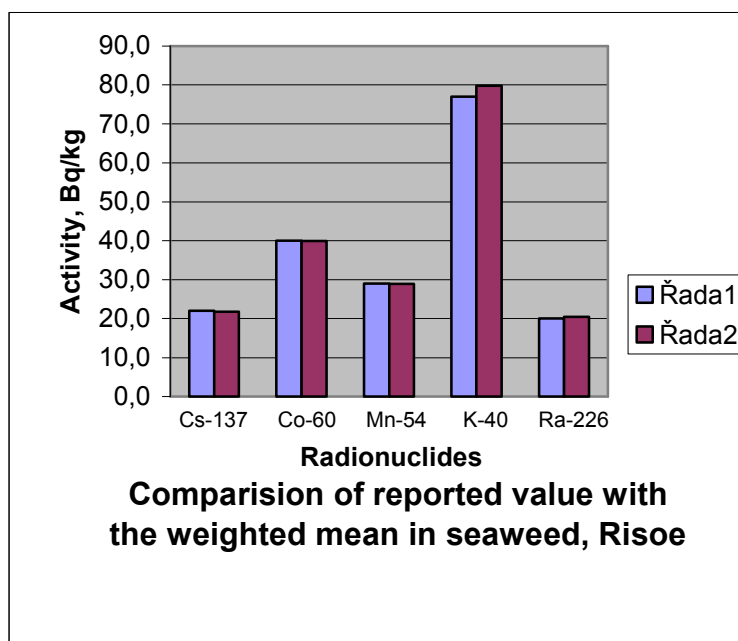


Fig. 1. Comparison of the laboratory's reported values for seaweed with the statistically weighted means of all participants accepted results.

Every year the accreditation body is performing the supervision of the laboratory. In June, 2003 the laboratory was accredited according to the ISO/IEC 17025 standard. This made it possible to participate and win in the tender for the 3 km zone monitoring control of the shutting down Salaspils nuclear reactor according to the Control Programme of State Significant Ionization Radiation Objects. The testing of 113 soil upper layers and 34 surface water samples was carried out.

### Conclusions

The 4 year work period on the quality control and quality assurance of the laboratory and the participation in the intercalibration measurements allowed

- to get the laboratory accreditation according to ISO/IEC 17025 standard;
- to enlarge the sphere of activities (introducing new methods);
- to increase the number of clients;
- to increase the number of tested samples from 124 in 2000 to 600 in the 8 months of the year 2003.

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3. Intercomparison of Radionuclides in Environmental Samples 2000-2001, Riso National Laboratory, Denmark, July 2002.



# **ANALYSIS OF SCRM EXPERIENCE IN THE AREA OF QUALITY ASSURANCE FOR RETROSPECTIVE EPR DOSIMETRY TECHNIQUE WITH TEETH**

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## **Introduction**

EPR dosimetry with tooth enamel is commonly accepted as one of most precise and accurate methods for retrospective dosimetry. At the same time, regularly conducted international Intercomparisons and Intercalibrations of EPR dosimetry techniques demonstrate the significant scatter of results among laboratories operation in this area. This is mainly caused by the lack of commonly adopted unified technique which would be based on clear and efficient scheme of quality assurance. In the present work we will summarize more than ten-year experience of Scientific Center for Radiation Medicine in the area of EPR dosimetry from the point of view of quality assurance.

It is commonly agreed that EPR dosimetry with teeth consists of following main steps:

- separation of enamel and sample preparation,
- registration of EPR spectra,
- spectra decomposition and evaluation of dosimetric signal magnitude,
- calibration of dosimetric signal in terms of absorbed dose,
- determination of accidental dose component,
- estimation of cumulative uncertainty.

For any particular version of EPR technique these steps may be characterized by specific set of factors influencing accuracy and reliability of dose reconstruction. The factors that are most significant in the SCRM technique are analyzed below.

## **Separation of enamel and sample preparation**

This stage strongly affects the performance of the whole EPR protocol and needs to be applied with caution. First, some EPR signals interfering with dosimetric signal of enamel may arise when mechanical forces are applied to the sample (Desrosiers, 1989). Then,

radiation sensitivity of enamel may change (Sholom et al., 1998). Finally, the treatment of enamel may significantly affect concentration of useful paramagnetic centers.

SCRM sample preparation procedure is aimed to minimize change of dosimetric signal. This is achieved through use of combined mechanical-chemical technology. At the first step of preparation, 2-3 mm layers, which consist mainly of pure enamel with small quantity of dentin, are cut from lingual and buccal sides of the tooth using low speed diamond saw. At the second step, the samples are ground to particles of 2-3 mm and treated in ultrasound bath in 2-N NaOH solution for several hours. The influence of such mild procedure of sample preparation on accuracy and reliability of EPR dosimetry technique has been experimentally tested and was found to contribute 10-15 mGy of additional error.

### **EPR spectra registration**

Slowly-changing EPR signals in the spectra of empty sample tube (Hayes et al., 1998) have effect on the results of spectra registration. The influence of this factor depends on model and configuration of EPR spectrometer and may be estimated via periodical recording of empty sample tube spectra (ESTS). In particular, for a spectrometer model BRUKER ECS 106 with cavity ER 4108 (model used in SCRM) the variability of signals in ESTS during one day may result in an additional error up to 50-100 mGy. Residual anisotropy of enamel samples contributes to uncertainty of spectra registration too. Additional error of several percents may be caused by possible drift of EPR spectrometer gain, about 1%.

Scheme of quality assurance for this step envisages daily tests of ESTS signals. It is acceptable if intensities of lines around g-factor of 2.0 in the difference of two ESTS (one recorded before and another after spectra of sample) have magnitude within 20 mGy as expressed in terms of dose. Influence of residual anisotropy is controlled estimating the difference between dosimetric signal in studied sample spectra and some reference spectra. Usually this difference is within few percents when the of sample tube goniometry is used.

### **Spectra decomposition and evaluation of dosimetric signal magnitude**

This step is one of most critical in EPR dosimetry technique due to possibility of false detection of EPR signals, especially in the case when several signals are overlapping (Grün, 2002). SCRM technique uses the decomposition procedure which is based on matrix method (Sholom and Chumak, 2003). In fact, application of advanced procedures of sample preparation and spectra registration, allow consideration of only 3 spectra for decomposition, reducing thus the probability of false detection of EPR signals. The quality control at this step

is expressed in estimation of differential signals in spectra between initial sample spectra and fitting spectra (sum of decomposition lines). Signals in this difference spectrum should be within 20-25 mGy.

### **Calibration of dosimetric signal in terms of dose**

This procedure is performed either by means of individual calibration of samples or using universal calibration curve for radiation sensitivity of enamel. The first option is considered as preferable in SCRM technique. The samples are irradiated on  $^{137}\text{Cs}$  source, which is calibrated against primary standard in NIST to precision of 3 % ( $2\sigma$ ).

Quality control for given step of dosimetry technique is implemented via use of the same irradiation conditions for all samples and periodic check of the calibrator.

### **Determination of accidental component of dose**

This step is one of most ambiguous in EPR dosimetry technique. Besides the transient (accidental) component of dose, three other components may contribute to cumulative dose of enamel, namely: background environmental exposure, x-ray diagnostic procedures and UV part of solar radiation. Often these components may be as high and even exceed significantly the value of accidental dose complicating this makes its correct determination.

SCRM technique use following approaches for correct accounting of non-accidental components (Chumak et al., 1999, Sholom et al., 2000). Dose component due to background environmental exposure is calculated as product of tooth age by annual dose rate at the area where tooth donor lives. Doses from x-ray diagnostic procedures are estimated by comparison of doses to lingual and buccal parts of tooth (Sholom et al., 2002). Due to unknown contribution from UV solar radiation, front teeth are excluded from routine dose assessment.

Quality control of these factors consists of following actions. The presence of data about liquidator's age and type of extracted tooth is checked in the tooth ID form (supplied by surgeon dentist). In the case when data is absent, the required information is received directly from liquidator. Then, the gradient of dose between lingual and buccal parts is checked, and depending on magnitude of this gradient, appropriate algorithm of accidental dose evaluation is applied (Sholom et al., 2002).

### **Estimation of cumulative uncertainty**

Objective verification of cumulative uncertainty for EPR dosimetry technique is achieved by international intercomparisons. Since 1995, SCRM took part in 6 such actions

(see Table 1 for details). Results demonstrated by SCRM in these intercomparisons are shown in last column of Table 1 in the following way. The mean absolute deviation of EPR doses from corresponding nominal values was calculated for all nominal doses below 300 mGy. For nominal doses higher than 300 mGy, the mean deviation of two sets of doses is expressed in relative units. So, according to Table 1, the mean error of SCRM technique calculated using all intercomparisons is 21 mGy for doses below 300 mGy and 11 % for dose higher than 300 mGy.

Table 1. SCRM results demonstrated during 1995-2003 in different intercomparisons of EPR dosimetry techniques

Title of intercomparison/ intercalibration	Year	Range of nominal doses (ND), mGy	Results of SCRM (mean deviation from ND: mGy for dose below 300 mGy/ % for doses higher than 300 mGy)
1 <sup>st</sup> International intercomparison	1995	0-1000	24 mGy / 5%
1 <sup>st</sup> Bilateral intercomparison with Utah University (USA)	1996	171-256	55 mGy
2 <sup>nd</sup> International intercomparison	1999-2000	99-815	2 mGy / 26%
2 <sup>nd</sup> Bilateral intercomparison with Utah University (USA)	2000-2001	74-810	20 mGy / 8%
Bilateral intercomparison with NIST (USA)	2001-2002	0-269	15 mGy
3 <sup>rd</sup> International intercomparison	2003	79-704	13 mGy / 15%
Mean values for all intercomparisons			21 mGy / 11%

## Conclusions

1. EPR dosimetry technique with teeth, which was developed and being used in SCRM for routine dosimetry of liquidators is characterized by two-level system of quality assurance. In-house level covers all steps of the technique and allows control and minimizing of uncertainties that arise on separate steps. Extramural level provides the control of reliability and accuracy of technique in whole by means of regular participation in bi- and multi-lateral intercomparisons.

2. Cumulative uncertainty ( $1\sigma$ ) of EPR dosimetry technique determined based on the results of 6 different intercomparisons is 21 mGy for dose below 300 and 11% for dose higher than 300 mGy.

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# **COST-BENEFIT OF VENTILATION AND AVERTED RADON IN DWELLINGS**

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## **Introduction**

The concentrations of radon ( $^{222}\text{Rn}$ ) and its decay products in the indoor air of dwellings depend mainly on their in- and outflow to and from the rooms. For the parent substance of radon the only significant removal process is ventilation while for its short-lived decay products ( $^{218}\text{Po}$ ,  $^{214}\text{Pb}$ ,  $^{214}\text{Bi}$  and  $^{214}\text{Po}$ ) in their various species the removal rates are determined – in addition to the radioactive decay – by ventilation, attachment to aerosol particles and deposition and resuspension on the walls and other surfaces [1]. These removal processes reduce the concentrations of radon daughters and thus diminish the inhalation dose. In spite of the complex processes in most of the cases the inhalation dose could be reduced significantly by increasing ventilation. On the other hand intensive ventilation causes monetary losses during the heating season. To assess an economically optimal ventilation rate we have introduced a cost-benefit analysis taking into account the cost of heating and benefit of averted dose due to ventilation.

## **Methods**

The cost of ventilation was calculated from the heating cost of the inflow air in the dwellings. Therefore the cost is the product of the rate of ventilation and the heating cost of unit volume of air to the proper temperature. Probably it's an underestimation of the real cost because additional heat loss might be provided due to heat conductivity of the walls and other structure elements of the building. Analyses have been provided by use of temperatures in- and outside, specific costs of the energy and other parameters represented for the Hungarian conditions.

The benefit of ventilation was assessed by the averted inhalation dose due to the mitigated radon concentration multiplied by the  $\alpha$ -value used as monetary value of unit-averted dose [2]. According to our former results [3]  $\alpha=10 \text{ €(person mSv)}^{-1}$  and the range of 1 standard deviation (65 % confidence interval) takes (2.1 – 52)  $\text{€(person mSv)}^{-1}$ . Taken into consideration the individual dose dependency by a power function the aversion coefficient is  $a=2.51 \text{ } 0.56$ . Both parameters were drawn among the public by the method of Willingness To Pay. To assess the collective dose the average number of persons in the dwellings was 3.

The model for simulation of the kinetics of  $^{222}\text{Rn}$  and its progenies (Fig. 1) contains the in- and outflows, binding process to aerosol particles, their deposition and resuspension to and from the surfaces (mainly walls). The mathematical representations are given by ordinary differential equations and the solution of them, including uncertainty analysis, were provided by the software ModelMaker [4]. For test calculations the binding rate coefficients were high ( $300 \text{ h}^{-1}$  for all progenies), the deposition of aerosols  $k_{\text{dep}}=0.8 \text{ h}^{-1}$  and the resuspension,  $k_{\text{res}}=0.5 \text{ h}^{-1}$  [1], the volume of the test room  $150 \text{ m}^3$ , the radon inflow (due to exhalation from ground and walls) 10 and  $20 \text{ kBq h}^{-1}$ . The outdoor concentrations of radon and all the daughters were negligible. The Equilibrium Equivalent Concentration (EEC) of radon was calculated by:

$$EEC(\text{Bq m}^{-3}) = 0.105 C_{\text{Po-218}}(\text{Bq m}^{-3}) + 0.515 C_{\text{Pb-214}}(\text{Bq m}^{-3}) + 0.380 C_{\text{Bi-214}}(\text{Bq m}^{-3}). \quad [1]$$

The average inhalation rate was  $1 \text{ m}^3 \text{ h}^{-1}$  and the dose coefficient  $7 \text{ nSv Bq}^{-1}$  [ICRP 65]. These parameters produce usually an annual inhalation dose of 3-10 mSv (relatively high).

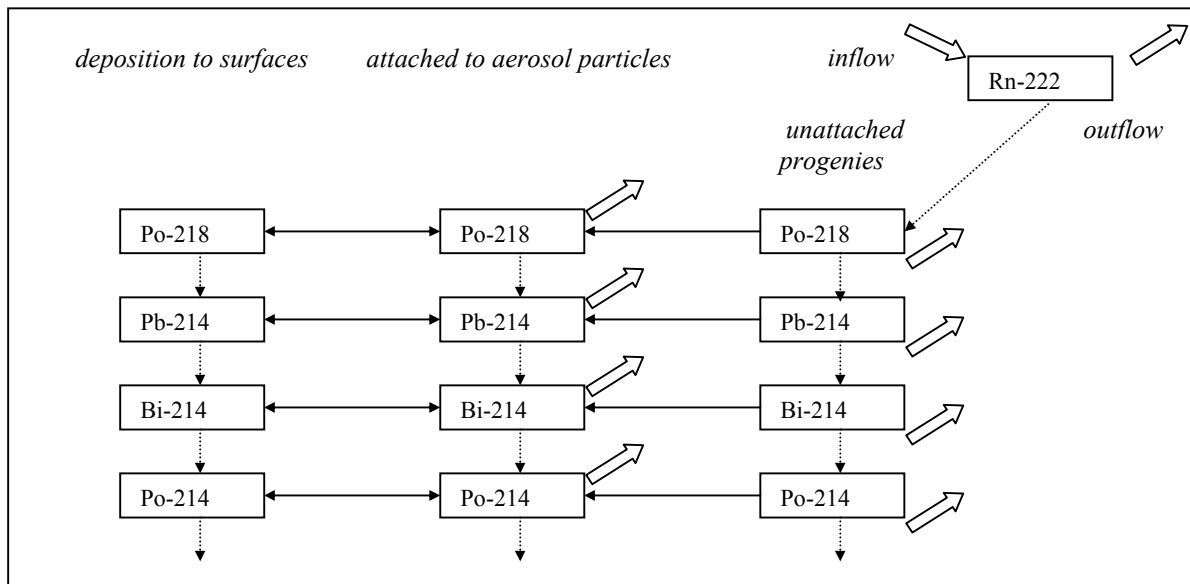


Fig. 1. Species and processes taken into account in modelling of the kinetics of short-lived radon daughters in dwellings (:.....→ radioactive decay, ⇔ : binding process, deposition and resuspension, ⇒: ventilation, in- and outflow)

## Results and discussion

Cost of heating and benefit of radon reduction due to increased ventilation were provided stepwise with respect to the rate of ventilation. Fig. 2. shows costs of a step of ventilation rate,  $\Delta S_v=7.5 \text{ m}^3\text{h}^{-1}$  (5 % of the total volume per hour) with respect to the total ventilation rate (full line). The cost of extra heating to compensate the increased ventilation by  $7.5 \text{ m}^3\text{h}^{-1}$  takes nearly 5 € and is constant with respect to the total ventilation rate (it depends only on the in- and outside temperature and other local conditions). But the benefit of constant averted dose - due to the step  $\Delta\lambda_v=7.5 \text{ m}^3\text{h}^{-1}$ - decreases nonlinearly with increased ventilation rate because the individual dose dependency of the  $\alpha$ -value is a power function. These relations are demonstrated in Fig. 2. for two different inflow rates of radon, by dashed lines. According to the results the optimal ventilation rates for the test-room are 15 and 35  $\text{m}^3\text{h}^{-1}$ , respectively.

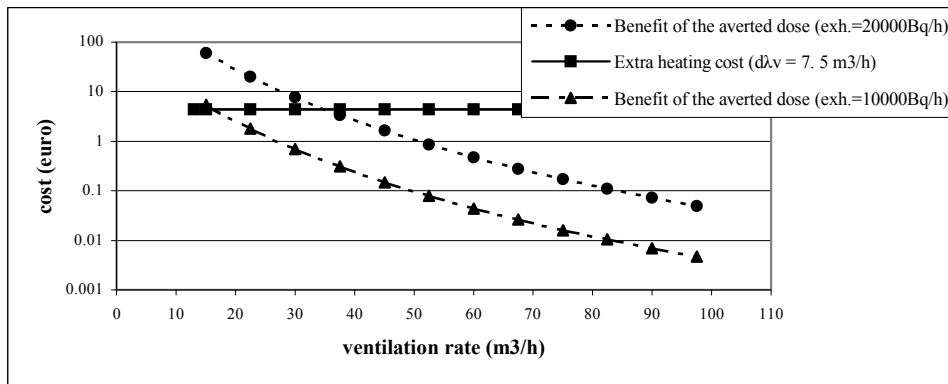


Fig. 2. Cost and benefit (€) of ventilation of a step modification by rate  $\Delta\lambda_v=7.5 \text{ m}^3\text{h}^{-1}$ , with respect to the total rate.

The former costs with respect to the radon activity concentrations in the test-room are shown in Fig. 3. By increasing radon concentration – due to decreasing ventilation - the individual dose and therefore the benefit of a constant averted dose is elevated sharply, meanwhile the cost of heating the air exchanged ( $\Delta\lambda_v=7.5 \text{ m}^3\text{h}^{-1}$ ) remained constant. According to the results the economically optimal radon concentration in room is  $250 \text{ Bqm}^{-3}$  for  $10 \text{ kBqh}^{-1}$  and  $550 \text{ Bqm}^{-3}$  for  $20 \text{ kBqh}^{-1}$  radon inflows. According to the uncertainty assessments the ranges of optimal radon concentrations might be varied by factors of 3-5, depending on the local parameter values. For better-defined condition the uncertainties are reduced.

In addition to the constant ventilation rate the periodically varied one was examined as well.



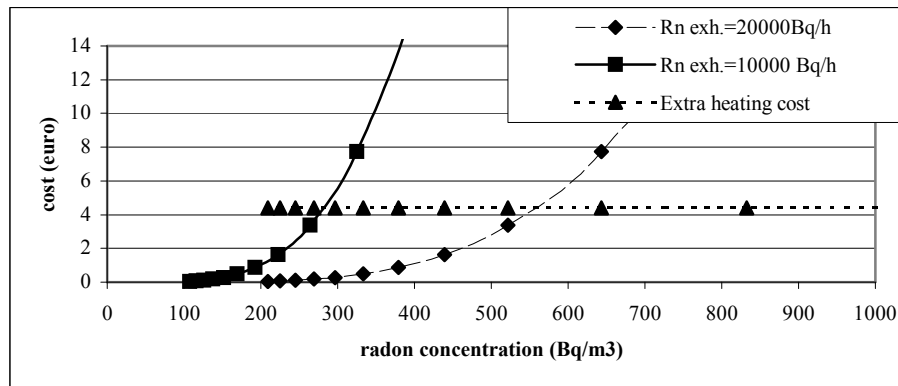


Fig. 3. Cost and benefit (€) of ventilation of a step modification by rate  $\Delta\lambda_v = 7.5 \text{ m}^3 \cdot \text{h}^{-1}$ , with respect to the radon concentration in the test room.

The example scenario for a period of 3 hours during daytime (including 10 minutes ventilation by exchange the total air volume) and no ventilation during night (8 hours) produces the same heating cost as the constant ventilation during the whole day, but the daily inhalation dose assessed is 0.00256 mSv for the periodic case meanwhile for the constant one 0.00420 mSv, in case of  $20 \text{ kBq} \cdot \text{h}^{-1}$  inflow of radon. It means the periodic ventilation has provided better dose relations than the constant one. Qualitatively it can be interpreted by the delayed built up of daughter concentrations in the air during periodic ventilation.

## Conclusions

The cost of heating due to the elevated ventilation for mitigation of radon content in dwellings can be compensated by the monetary benefit of the averted dose, in case of higher (annually 3-10 mSv) exposure. During the heating season the economically optimal ventilation takes  $0.1\text{-}0.5 \text{ h}^{-1}$ , meanwhile the radon concentration in the indoor air decreases to  $200\text{-}800 \text{ Bq} \cdot \text{m}^{-3}$ , depending on the exhalation of radon, number of persons living in the dwellings and other local parameters. Our results from the optimal planning correspond to the radon concentrations recommended by the international organizations as action levels. In general, the periodic ventilation in daytime provides a higher averted dose than the constant one in case of the same heating cost.

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# VISUALIZATION OF EXCEEDING THRESHOLD PARAMETERS IN THE GAMMA CAMERA FOR QUALITY ASSURANCE

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## ***Introduction***

The quality of gamma cameras should be checked regularly according to the guidelines of ÖNORM S 5271 in order to detect deviances of the machines. Whereas for some of the measured parameters threshold limits exist, there are only few images for visualisation of the loss in image quality available (Busemann et al [1]).

## ***Material and Methode***

The conceptual formulation was to observe the loss of quality while simulating deviances of the gamma camera. In this work 2 different machines are compared:

Elscont Helix, Serial: 2408, 2 head system with a parallel-whole colimator, HPC-46  
[LEUHR]

Elscont SP-4M, Serial : 2302, 1 head, with a parallel-whole collimator APC-34S  
[LEAP]

In order to evaluate the results a XPertPro- Workstation with filtered back-projection was used.

For the first step we concentrate our work on the results of the quality control, where a flood-phantom was filled with a radioactive source (Tc-99). Pictures with and without all corrections (NOLES) have been taken with the parallel-whole collimator HPC-46. This was

done to be able to compare the results after changing to the orthogonal-whole phantom (OHP), which was evaluated with and without linearity corrections (NOL).

In the second step the same procedure was done with a Jaszczak-phantom, which is made of plexi glass in order to check the image quality of a SPECT System. A Jaszczak-phantom is a cylinder (diameter 22 cm, height 16 cm) where spheres and rods with different diameters can be placed at certain positions.



Fig 1. Flood phantom, Jaszczak-phantom.

### ***Results***

The difference with and without automatic correction with respect to their homogeneity can be seen in the following figures:

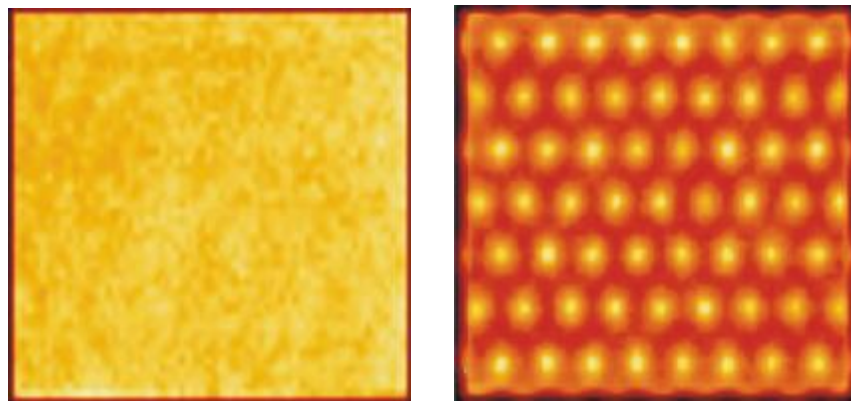


Fig 2. Picture with and without corrections (NOLES); the head of the rectangular gamma camera shows a homogeneity of  $\sim 4.5\%$  and  $\sim 27.7\%$ .

The orthogonal-hole phantom was chosen in order to compare the results with and without linear correction (NOL).

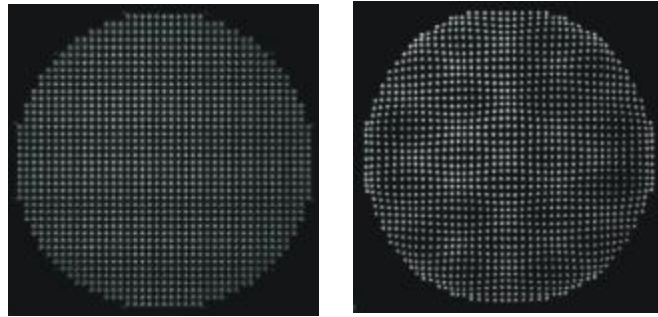


Fig. 3: The orthogonal-hole phantom measured with the circular camera with and without linearity correction.

In the following images of the orthogonal-hole phantom with various activity concentrations of Tc-99m were acquired to show the effect of dead-time losses on image quality. Comparing all photomultipliers (PM) adjusted and one centered PM non-adjusted, a point source of Tc-99 was placed in the distance of 5 times the diameter of the camera. The intrinsic inhomogeneity, was measured in the image being taken with adjusted PMs (fig. 4, left) to 3.5 % and in the non- adjusted image, (fig. 5, right) to 9.3 %.

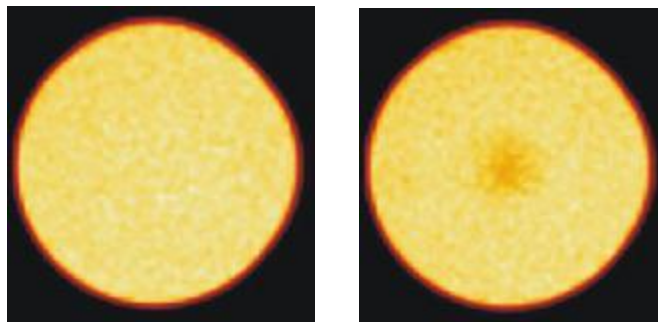


Fig. 4: The intrinsic inhomogeneity of the head of a SP-4M gammacamera, left: all PMs adjusted, right one, centered PM non-adjusted.

The influence of the amplification gain for an adjusted and non-adjusted photomultiplier results in differences in the energy of the photo-peak and the full width at half maximum, which is documented in table 1.

	Energyspectrum photomultiplier		Energyspectrum head of the camera	
	Adjusted	Non-adjusted	Adjusted	Non-adjusted
FWHM	8.88 %	8.77 %	10.16 %	9.89 %
Photopeak-energy	139.58 keV	134.27 keV	143.07 keV	142.42 keV

Table 1: Adjusted and non-adjusted photomultiplier of the energy spectrum of the photomultiplier and the head of the SP-4M camera.

In order to evaluate the image quality of a SPECT System with planar inhomogeneities the Jaszczak-phantom was filled with Tc-99 and a SPECT acquisition was performed (planar inhomogeneity 5.3 %). For a second acquisition a line-source was placed near the center of one head of the camera. This lead to a planar inhomogeneity of 10.6 %. Reconstruction algorithm amplified these planar inhomogeneities to SPECT inhomogeneities of 5.9 % and 23.1 % as can be seen in fig. 5.

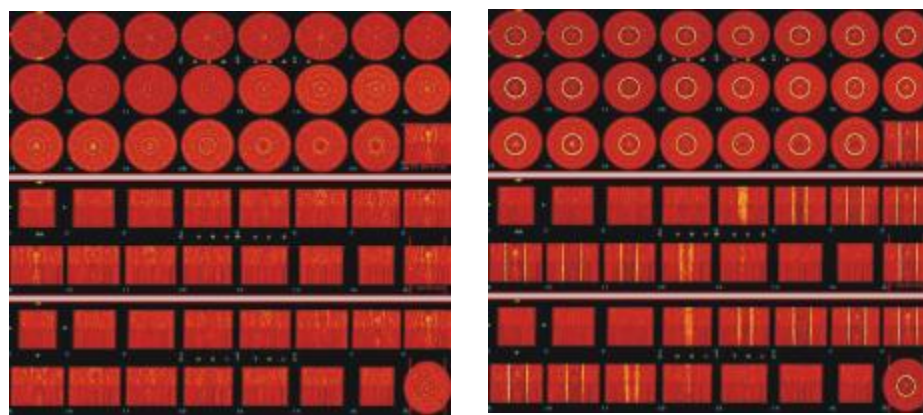


Fig. 5: Jaszczak-phantom, both pictures are filtered back-projections (RAMP) and corrected for attenuation; left: homogeneity 5.9 %, right: line source at 10 cm from the center, homogeneity 23.1 %.

## Discussion

The quality control in Nuclear Medicine is important because small artefacts can play an important roll in diagnostics. It is therefore necessary to get an idea of how pictures with bad quality parameters look like. The intensive deal with software parameters, like the linearity, the energy or the sensitivity correction is done according to machine requirements. Therefore it is necessary to be informed about the maximum in deviation, which can occur in between.

[1] E. Busemann Sokole, L.S. Graham, A. Todd-Pokropek, A. Wegst, C.C. Robilotta, A Quality Dept. Nuclear Medicine, Academic Medical Centre, Amsterdam, The Netherlands, STI/PUB/1141, 293 pp.; 2003, ISBN 92-0-101303-5.