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# IM 2022
# NEUDOS 14

**25-29.04.2022**
**Kraków, Poland**

# BOOK OF ABSTRACTS

# International Conference on Individual Monitoring of Ionising Radiation (IM2022) and Neutron and Ion Dosimetry Symposium (NEUDOS-14)

**Organized by**
**The Henryk Niewodniczński Institute of Nuclear Physics Polish Academy of Sciences**
**The European Radiation Dosimetry Group**

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*Auditorium Maximum of the Jagiellonian University,*
*ul. Krupnicza 33, Krakow, Poland*
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*Krakow, 2022*

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# The European Radiation Dosimetry Group – A 40 Year Success Story

Werner Rühm, Helmut Schuhmacher, Pascal Pihet

## Abstract

The European Radiation Dosimetry Group (EURADOS) was founded in 1981/82 to foster collaboration in the dosimetry of ionising radiation. Since then the group has continuously developed and is currently a network of 80 institutions and more than 600 individual scientists supporting research and development of dosimetry and harmonisingdosimetric practice across Europe, including also exchanges with the scientific community outside of Europe. This paper describes the various steps taken and strategic decisions made by EURADOS members over the years, from the very beginning when the idea was born, via periods during which the role and strategy of the network had to be defined, elaborated, and refined. These steps and decisions paved the way for EURADOS as it appears today. In addition, the paper sketches more recent actions that were taken to form an independent self-sustainable association which is a reliable partner for various international organisations working or being interested in radiation research and radiation protection. Major activities of EURADOS are highlighted such as a) organisingdosimetricintercomparisons on a regular basis for quality assurance of dosimetry procedures, b) establishing Working Groups whose scientific achievements are published in the peer-reviewed scientific literature, c) developing and offering education and training events, and d) contributing towards the development of strategic and integrated radiation research in Europe. The network is a living organisation which proved its flexibility by continuing its operations during the Covid-19 pandemic without any disruption. We conclude that EURADOS is fit to follow its mission in the future and are confident that the network will successfully master all challenges that might lay ahead before the 50$^{th}$ anniversary will be celebrated.

# Individual monitoring of the public after radiological emergency: evolution over the last three decades

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Although individual dosimetric monitoring of the members of the public is not envisaged by any of preparedness plans, its application after Chernobyl proved to be justified. In 1987-1997, doses to the members of public were monitored using automated TLD systems. The dosemeters were distributed among various age, professional and social groups of the population of urban and rural areas of Ukraine downwind Chernobyl. The results of individual monitoring served to evaluation of the parameters of ecological dosimetric models (e.g. behavior factors for various groups of exposed population), validation of the results of dosimetric passportization and reassurance of the public in validity of official dose estimates. These measurements were terminated in 1997 due to decline of doses due to Chernobyl release in the populated areas below the level of natural background.

Similar experience was repeated in Japan after the Fukushima accident when passive (glass) and active (D-Shuttle with semiconductor detector) personal dosemeters were distributed among the residents of contaminated areas, predominantly – pregnant women, school children, and adults. The latter technology allowed to record hourly dose values and estimate individual cumulative doses of gamma exposure over the period of up to one year.

Fukushima accident was marked by appearance of a new phenomenon – self organized non-professional dose rate measurements performed by a citizen science network called Safecast. Further development of this citizen science approach is expressed in the use of specially designed and improvised personal dosemeters/dose rate meters based on smart phones with internet access and CCD cameras, which can be used as ionizing radiation detectors.

The EU-CONCERT funded SHAMISEN SINGS (Nuclear Emergency Situations: Improvement of dosimetric, Medical And Health Surveillance - Stakeholder INvolvement in Generating Science) project aims to explore how citizen science can be promoted for dose measurements and information dissemination through the use of mobile Apps that can be made widely available. This is particularly important at the initial phase of radiological emergency when regular highly professional measurements are not available yet.

So, over the time since Chernobyl, individual dose monitoring of the general public went all the way from some unique practice to a wide scale citizen science application area. The talk will discuss pros and cons of each of the considered methodologies as well as the limits of their application.

# Experiences in Transitioning a Testing Laboratory to the New ISO/IEC 17025:2017 Standard

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The international standard ISO/IEC 17025 “General requirements for the competence of testing and calibration laboratories” was first issued in 1999 by the International Organization for Standardization (ISO) and the International Electrotechnical Commission (IEC). It is regarded as the global reference for laboratories wanting to demonstrate their capacity to operate competently and deliver valid results, and often used as the basis for accreditation. Since its inception, the standard has been revised twice, in May 2005 and most recently in November 2017. This third revision contains some significant changes, for which laboratories and accreditation bodies alike had to prepare. While each accreditation body may determine its own policy on how to handle the transition to the new revision of the standard, all accredited laboratories were required to complete the transition by 1 December 2020. The most substantial changes are as follows: (i) adoption of a new structure to align the standard with other existing ISO/IEC conformity assessment and quality management standards; (ii) introduction of the concept of risk-based thinking, highlighting the commonalities with the new version of ISO 9001 “Quality management systems – Requirements” and strengthening the requirements for impartiality and confidentiality; and (iii) harmonization of terminology with the International Vocabulary of Metrology (VIM). Prescriptive requirements were reduced in number and largely replaced with performance-based requirements, allowing for greater flexibility in processes, procedures, documented information, and organizational responsibilities. This paper disseminates the experience of the IAEA Radiation Safety Technical Services Laboratory in successful transition to ISO/IEC 17025:2017 and re-accreditation after its third surveillance cycle. It covers all stages from deciding on the overall timeline for transition, conducting a gap analysis between the existing quality system and the requirements in the revised standard, updating the quality management system documentation, creating training and communication plans for laboratory personnel, to monitoring the changes and improving the system through auditing, management review and participation in proficiency testing schemes. Lessons learnt with regard to building operational resilience and maintaining a business continuity management system to prepare for, respond to, and recover from disruptions are considered.

# Revision of the series ISO 6980: reference beta-particle radiation

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**Keywords:** Beta radiation fields, ISO standard 6980, revision

Since 2019 the ISO working group on “Reference radiation fields”, ISO TC85 SC2 WG2, is revising the ISO series 6980, of all three parts in parallel. The following changes are being considered:

- alignment to ISO 29661:2012 and its Amd 1:2015;
- differentiation between the conversion coefficients for $H_p(0.07)$ and $H'(0.07)$;
- inclusion of the quantities $H_p(3)$ and $H'(3)$;
- inclusion of the small photon component in the reported dose(rate) as it contributes to the total dose and may influence the instrument’s indication significantly;
- inclusion of typical and up to date electron and photon spectra;
- harmonization of the ISO water slab phantom substitute in all parts to a ≥ 20 x 20 x 2 $cm^3$ plate made of polymethyl methacrylate (PMMA);
- inclusion of two new radiation fields with a mean energy between the ones of $^{85}Kr$ and $^{90}Sr/^{90}Y$ (based on $^{90}Sr/^{90}Y$ behind thin absorbers made of a PMMA);
- inclusion of correction factors for additional combinations of the distance from the radiation source to the irradiated object and the presence of the beam flattening filter, especially for $^{106}Ru/^{106}Rh$;
- inclusion of the Spencer-Attix theory besides the Bragg-Gray theory;
- removal of $^{14}C$ ($E_{beta,max}$ = 0.16 MeV) from the complete standard series.

Currently (November 2021), the draft international standards (DIS) are being prepared. The final standards are expected to be published in mid-2023.

The author wishes to thank the members of ISO TC85 SC2 WG2 and other contributing colleagues from all over the world.

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# Microdosimetric modeling of radiation-induced effects in detectors and living organisms: what can these two worlds learn from each other?

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The effect of radiation on detectors and living organisms strongly depends on both the exposure characteristics (i.e. dose, dose rate, particle type, LET, radiation quality, fractionation) and the system's ability to react to the stimulation. It is generally accepted that predictive models dealing with systems characterized by sensitive targets in the order of micro- and nano- meters (i.e. DNA, chromatin loops, luminescent centers) must account for the stochastic behaviour of the energy deposition at these scales. The study of the stochastic nature of the energy deposition takes the name of microdosimetry [1] and was developed in the '50s to overcome the limitations of average quantities such as absorbed dose and the linear energy transfer (LET) in describing the radiation effects in biological samples. Since then, many microdosimetry-based approaches were developed and used in the fields of radiation biophysics and detector science.

In this talk, a mathematical parallelism is established between existing approaches to model the response of different radiation detectors and biological organisms such as mammalian cells, viruses and bacteria.

This includes:
1) The description of the macroscopic dose-response through multi-hit theories, the linear-quadratic model, and other approaches.
2) The similarity between the particle-dependent ion-vs-LET trends of relative quantities used to describe the effect of changes in radiation quality such as the relative biological effectiveness (RBE) and the relative detector efficiency.
3) The correlation between the time-dependent ability of the system to recover from the radiation stimulation (i.e., DNA repair mechanisms, competing luminescence centers and detector signal fading) and the two aforementioned system characteristics (dose response and radiation-quality dependence).

As in Olko's seminal work [2], the comparison between the different response/biological-weighting functions was facilitated by converting the relevant quantities into water-equivalent ones. The radiation transport code PHITS [3] and its microdosimetric function were used for this purpose. Relevant similarities in the results were found and will be discussed.

Although attempts were made in the past to relate the fields of radiation-action modeling for detectors and living organisms, these findings strongly suggest that the two scientific communities would greatly benefit from a stronger interconnection.

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# Verification of microdosimetric biological weighting function for the $RBE_{10}$ modelling in particle therapy using solid state microdosimeter

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**Introduction:** With the growing interest and use of charged particle therapy for cancer treatment [1], it is becoming increasingly important to correctly predict and quantify the relative biological effectiveness (RBE) for treatment. The modified microdosimetric kinetic model (MKM) [2] and the biological weighting functions (BWFs) [3] being the two most commonly used for RBE prediction and utilizing a microdosimetric approach. Due to shortcomings in the BWF model, a new model has been proposed in [4], namely the improved biological weighting function (IBWF) - r(y), which is convoluted with the microdosimetric lineal energy spectra for a range of ions spanning from $^{1}H$ to $^{238}U$ to predict $RBE_{10}$. In this study, we compare the $RBE_{10}$-vs-dose average lineal energy ($y_D$) trends obtained by the IBWF model with simulated and experimentally obtained microdosimetric quantities using a silicon-on-insulator (SOI) microdosimeter developed by the Centre for Medical and Radiation Physics and compare with the $RBE_{10}$ predicted by the modified MKM.
**Methods:** $RBE_{10}$ and $y_D$ were found from the microdosimetric spectra for $^{1}H$, $^{4}He$, $^{7}Li$, $^{12}C$, $^{14}N$, $^{16}O$, $^{20}Ne$, $^{28}Si$, $^{56}Fe$ and $^{131}Xe$ ions with energies from 5 to 500 MeV/u simulated with the Monte Carlo particle and heavy ion transport code system (PHITS) taking into account the target volume is 1 μm diameter water sphere [5]. A correlation was then found between the simulated results, and $RBE_{10}$ values for V79 cells irradiated with monoenergetic ions obtained through the Particle Irradiation Data Ensemble (PIDE) [6]. These results are then compared to experimentally derived $RBE_{10}$ values obtained with the SOI microdosimeter utilizing the modified MKM with V79 cell parameters. Experiments were carried out at a number of proton and heavy ion therapy facilities with various ions and energies.
**Results:** It has been shown that the modified MKM predicted $RBE_{10}$ based on experimental microdosimetric spectra is in good agreement with IBWF model for $^{4}He$, $^{7}Li$, $^{12}C$, $^{14}N$, $^{16}O$, $^{20}Ne$, $^{28}Si$, $^{56}Fe$ and $^{131}Xe$ ions within experimental error including overkill region. For protons, the modified MKM predicted $RBE_{10}$ values are less than the IBWF model for used V79 parameters that require further investigation. Full results will be presented in the full paper.
**Conclusions:** It has been shown that the SOI microdosimeter is capable of experimental RBE prediction and verification while being in agreement with the IBWF model ultilising simulated microdosimetric spectra with 1 μm water sphere and V79 cell survival data.

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# A novel hybrid approach for radiation field characterization: from detector development to biological damage modelization

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Microdosimetry allows the investigation of damages induced by radiation in a micrometer-size volume, and it has been exploited in particle therapy applications to accurately characterize the radiation field quality.
One key microdosimetric quantity is the lineal energy y, which is defined as the energy $\varepsilon$ deposited by an event inside the detector divided by the mean path length *l*, which is only a geometrical parameter.
We have designed a new detector *Hybrid Detector for Microdosimetry* (HDM) [1], composed of a tissue equivalent proportional counter (TEPC) microdosimeter followed by 4-layers of low gain avalanche detectors (LGADs). This design provides a direct measurement of the energy imparted in tissue provided by the TEPC, as well as particle tracking with a submillimeter spatial resolution given by the LGADs.
Together with measuring microdosimetric y distributions, with HDM we can measure a physical quantity that we referred to as $y_T$, equal to the energy deposited by radiation in the TEPC divided by the real path length obtained with the LGADs. Adding a tracker to the TEPC can help improving the characterization of the radiation field quality, for example measuring the zero events (i.e. particles that traverse the microdosimeter without depositing energy).
Using Geant4 toolkit, we investigated HDM performances in terms of detection and tracking efficiencies when placed in water and exposed to protons and carbon ions in the therapeutic energy range. We applied modern machine learning techniques to improve tracking efficiency, especially for minimizing the loss of particles due to scattering.
We also explored the use of $y_T$ distributions measured with HDM for predicting cell survival and RBE, and developed a fully probabilistic microdosimetry based model (Generalized Stochastic Microdosimetric Model $GSM^2$) [2].
The results of both HDM and $GSM^2$ studies will be presented, focusing on the advances that this novel real track length-based approach can bring to particle therapy. In particular, we will demonstrate the feasibility and advantages of the proposed detector design, and show that machine learning can be used to recover lost information and reach a 100% tracking efficiency. We will also discuss how describing the radiation quality with $y_T$ spectra can impact the accuracy of radiobiological models for cell survival and RBE.

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# Correlations between energy imparted and ionization yield at the nanometre scale


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**Purpose:** In microdosimetry, measurements of the energy imparted $\varepsilon$ are generally carried out through a measurement of ionizations, based on the assumption that $\varepsilon$ is equal to the number of ionizations $\nu$ multiplied by the conventional W-value, the mean energy expended per ion pair formed when the initial kinetic energy of a charged particle is completely dissipated in the medium. However, when the number of interactions is less than approximately 10, this assumption fails [1]. This work aims to study if another conversion factor $\tilde{\omega}$ can be used so that the discrete distribution of $\varepsilon$, *P*($\tilde{\omega}\cdot\nu$), where $\varepsilon$ is calculated multiplying the number of ionization $\nu$ by $\tilde{\omega}$, has at least the same mean value (corresponding to $\bar{y}_F$) and the same ratio of the second to the first moments ($\bar{y}_D$) as the *f*($\varepsilon$) distribution, where $\varepsilon$ includes all the energy deposits. The assumption to use as a conversion factor the conventional W-value is relaxed, and the probability for events of zero-size is also accounted for.

**Methods:** Monte Carlo simulations were performed with the Geant4-DNA track structure code [2] in water spheres of different diameters. Primary beams of protons and carbon ions in the energy range 1 – 100 MeV/u were considered. Both the imparted energy $\varepsilon$, which includes all relevant interaction types, and the number of ionizations $\nu$ were scored simultaneously and their mean values calculated. The quotient of $\bar{\varepsilon}$ divided by $\bar{\nu}$ was calculated for all radiation qualities and different site sizes and defined as $\tilde{\omega}$. Variations of $\tilde{\omega}$ were studied as a function of volume size, particle type and energy.

**Results:** The $\tilde{\omega}$ value depends only weakly on particle type and energy, while stronger variations are observed with respect to sensitive volume size. A conversion factor can be used, which is almost independent of radiation quality but depends on *SV* size, to calculate the imparted energy as the product of $\tilde{\omega}$ and the number of ionizations $\nu$. The mean value of the imparted energy calculated in this way, starting from the distribution of the number of ionizations, is equal to the mean value obtained by scoring directly the imparted energy, with deviations smaller than 5%. A similar agreement was also found for the values of the second to first moments ratio.


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# A comprehensive Monte Carlo simulation of the neutron response of multi-element microdosimetric detectors based on thick gas electron multiplier

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We present comprehensive Monte Carlo simulations for the neutron response of the multi-element THick Gas Electron Multiplier (THGEM) microdosimetric detector. THGEM's absence of wire electrodes greatly simplified building multi-element detectors. A Monte Carlo simulation was developed using the MCNP6.2 code to calculate the deposited energy spectrum in the gaseous cavities for three different multi-element configurations, 7x3, 19x5, 37x7 that occupy a cylindrical volume of 5 cm diameter by 5 cm length. The neutron energy was varied from values 10 keV to 2 MeV and the angular response was also investigated. The simulated energy response showed a good agreement against the evaluated fluence-to-kerma conversion coefficients in the neutron energy region 10 keV to 100 keV while discrepancies were observed in the region above 250 keV. In order to find the source of the discrepancy, the responses of each layer and cavity were simulated. A calibration algorithm that can correct the discrepancy and enhance the dose equivalent response is currently under investigation.

# Microdosimetry with a tissue-equivalent proportional counter at the MedAustron light ion beam therapy facility

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**Aims**

The aim of this work is to establish an experimental procedure for the microdosimetric characterization of the radiation quality of ion beams in the specific clinical environment of the MedAustron therapy facility. The first microdosimetric spectra measured with a gas detector at the proton beams of MedAustron are presented and discussed.

**Materials and Methods**

Tissue-equivalent proportional counters (TEPCs) are the golden standard in microdosimetry. Therefore, a miniaturized TEPC (mini-TEPC) [1] was used to collect microdosimetric spectra at different positions along the depth dose profile of a monoenergetic 62.4 MeV single-spot proton beam with a full width at half maximum of approximately 2.5 cm. A reduced particle rate of approximately 4 MHz was maintained to minimize pile-up and saturation effects in the detector. The mini-TEPC has a cylindrical sensitive volume with equal diameter and height of 1 mm and is filled with low pressure propane gas to simulate a size of 1 μm at unit density.
The spectra obtained for the MedAustron beam were compared to those measured at other facilities at approximately the same nominal beam energy [2]. The comparison was performed at similar positions along the depth dose curve. Monte Carlo simulations have also been performed and benchmarked with experimental data. Simulations were used to study the features of the measured spectra in relation to the characteristics of the experimental set-up and the radiation field.

**Results and Conclusions**

In general, experimental spectra measured at different centres with approximately the same nominal beam energy show very similar shapes in the distal region of the depth dose profile. In contrast, some deviations can be seen in the entrance region, potentially reflecting differences in the radiation fields and/or detectors used. The feasibility of characterizing the radiation quality of the MedAustron proton beam has been demonstrated.

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# Microdosimetry of a 148 MeV modulated proton beam with a new mini-TEPC

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A new mini–TEPC with cylindrical sensitive volume of 1.0 mm in diameter and height, and with external diameter of 1.2 cm, has been developed to work without gas flow. This detector is the prototype of an engineered device, designed and realized with the aim of containing the production costs and assuring a better reproducibility of the machining processes. With this mini-TEPC we have performed the microdosimetric characterization of the 148 MeV modulated proton beam at the radiobiological research line of the Trento proton therapy centre [1]. The aim of this work was to study the capability of this mini-TEPC to monitor the radiation quality of the radiation field across the Spread Out Bragg peak (SOBP).

Measurements were performed at several positions along the SOBP, by inserting layers of solid water of different thicknesses in front of the detector. At four specific depths, the response function of the new microdosimeter has also been compared to that of a commercial spherical microdosimeter, the LET-1/2 Spherical Tissue Equivalent Proportional Counter produced by Far West Technology.

From the microdosimetric spectra the dose–averaged lineal energy values were derived and studied as a function of depth. Monte Carlo simulations both for the LET and the microdosimetric spectra were also performed and compared to experimental results.

Microdosimetric RBE has been calculated through the Loncol's biological weighting function [2] and used to calculate the biological dose, which was compared to the one obtained with a fixed RBE of 1.1.

The response of the new mini-TEPC is in good agreement with that of the commercial detector, and also with Monte Carlo simulations. The consideration of a variable RBE to weight the physical dose underlines an underestimation of the delivered biological dose beyond the Bragg Peak when the latter is weighted with a fixed RBE=1.1.

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# An analysis of occupational doses in the proton therapy center in Poland in 5 years of practice

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Proton therapy is the most rapidly developing technology in modern radiotherapy because of the unique properties of proton beams, which allow for highly conformal irradiation of the tumor volume.

The first proton radiotherapy facility operating in Poland was built at the Institute of Nuclear Physics Polish Academy of Sciences in Krakow. Since 2016 proton therapy is performed in the facility with isochronic cyclotron Proteus C-230 with the proton beam up to 230 MeV.

Radiological protection of medical staff at proton beam therapy facilities requires special attention. Appropriate management and control are required for the therapy equipment and also for the air in the treatment room which can be activated by the particle beam and its secondaries.

This paper will highlight important radiation safety issues related to proton therapy, and focus on occupational exposure for workers. Staff members were divided into 2 groups: radiation technologists with physics, and cyclotron support. The first group participates in clinical work with the patient or performs measurements quality control of patients' position and treatment plans, while the second supports accelerator (cyclotron).

In all measurements, thermoluminescence dosimetry was used. Low occupational doses for the whole body in terms of Hp(10) were measured for the radiation technologist and physicists. The highest measured dose didn't exceed 0.25 mSv/quarter (the annual limit of 20 mSv). While a group of cyclotron service employees receives doses at a higher level (doses above the natural background were measured for 23% of workers). Cyclotron operating personnel were also rung dosimeter to measure skin doses in Hp(0,07). Doses on the skin are measurable, however, lower by 2 orders of magnitude from the value of the limit doses 500 mSv per year for the skin.

Our results showed that the exposures of medical staff members were far below the appropriate annual dose limits. We concluded that the present legal controls and level of radiation protection in Cyclotron Centre Bronowice are on a good level and sufficient for staff members.

# Commissioning of the operational radiation protection in Compact Proton Therapy Centers (CPTC) summarized in ten recommendations

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The development and construction of a proton therapy center normally takes between two and three years to complete, nevertheless, radiation protection assessment measures should begin in the planning phase of the center and go along with the execution of the facility. There are a large and exhaustive studies about the implementation of radiological protection measures in proton therapy facilities, however, protontherapy discipline is constantly evolving and incorporating new developments that pose a great challenge for radiation protection of patients, medical staff, exposed workers, and the general public. The design of some aspects of operational radiation protection was developed from 2018 until now, within the research project Contributions to operational radiation protection and neutron dosimetry in compact proton therapy centers, including evaluation of shielding and activation in barriers, evaluation of ambient neutron monitors, comparison of neutron fields yielded with current and new proton delivery methods, among others, with both, experimental measurements and simulations with Monte Carlo codes as MCNP6.2. The aim of this work was to present a commissioning process of the operational radiation protection of Compact Proton Centers, summarized in ten main recommendations, achieved in the activities mentioned above, and lined up with the requirements of the Spanish Nuclear Authority and international guides. These recommendations include, for instance, verification that shielding has been executed in compliance with the prior authorization, compilation of certificate of materials of barriers, with composition of cement, and final density of concrete, ambient monitoring, and control system in operation, including the justification, limitation and optimization of actions propose. The goal of this process is to guarantee the compliance with the dose limits for clinical staff, technical staff, and public

**Keywords:** *Compact proton therapy centers, Operational radiation protection, Neutron dosimetry, Commissioning*

# A pulsed high-energy photon reference field for testing dosemeters used for radiation protection measurements behind shieldings from medical accelerators

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Radiation protection at modern accelerator facilities used in science, technology and medicine shall meet the legal and normative requirements of personal dosimetry.
Currently, dosimetric measurements at these facilities are performed with area dosemeters whose suitability for these high-energy and pulsed radiation fields cannot be tested according to the state of the art in science and technology due to the lack of suitable reference fields. Therefore, the establishment of reference radiation fields is necessary, which allow measurements traceable to national standards and tests for the development and verification of new dosimetry systems.
Currently, type testing is usually performed up to energies not larger than 7 MeV and in continuous radiation fields. The performance of radiation protection dosemeters, however, can be compromised by pulsed and high-energy radiation fields.
Well-characterized reference radiation fields are therefore necessary to ensure reliable measurement with such devices.
Within the project “Establishment and characterisation of a reference field for ensuring radiation protection at accelerator facilities in medicine and research and for testing and calibrating of corresponding measuring instruments” such a reference field will be set up. It will be a fully-characterized pulsed photon reference field resembling increased ambient dose levels such as those occurring behind (insufficient) shielding of medical and research accelerator facilities. This reference field could then be used for type testing dosemeters typically used for ambient or personal dosimetry.
A commercial medical linear accelerator at PTB was chosen as the radiation source. The shielding comprised a 2 m-thick wall of a concrete composition typically used for radiation shielding.
In this presentation results from the characterization will be shown, where: ambient dose rates behind the shielding (with respect to the beam direction) were conducted using a secondary standard ionization chamber to allow for traceability to national standards. Energy spectra behind the wall were determined by measurements and unfolding techniques with a passive few-channel spectrometer for pulsed photon fields and for neutrons PTB’s Bonner sphere spectrometer was applied. Additionally, Monte Carlo simulations of the energy spectra and dose distribution behind the wall were performed at various distances along beam axis.
The project is funded by the BfS (project no. 3619S2236).

# Study of the impact of the newly proposed ICRU/ICRP quantities on personal doses in a realistic radiation workplace: interventional radiology case study

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The International Commission on Radiation Units and Measurements (ICRU) proposed a new set of operational quantities for radiation protection for external radiation in its Report Committee 26 [1]. The new proposal aims to improve the coherence between the operational quantities and the definitions of the protection quantities in ICRP's 2007 Recommendations. The main difference between the present set of quantities is the redefinition of the operational quantities, which are no longer dependent on doses at specific points in the ICRU sphere and soft tissue, but will be based on particle fluence (or air kerma) and conversion coefficients. It is expected that this change in operational quantities will impact both dosimeters and reference radiation fields used for their calibration. Although for many energies the conversion coefficients will change relatively little, for radiation fields with low energy photons component, such as medical X-ray applications, there will be a significant decrease of the values of the conversion coefficients. This will mean that the numerical values of the new operational quantities will be much lower for the same radiation field. These values will be closer to the effective dose, but still this change can cause confusion for the workers. It is important to examine the effect of the new set of dose conversion coefficients on the personal dose in realistic radiation fields. Till now, no study has been performed to quantify the effect of switching to the new quantities for realistic scatter fields in interventional radiology.

We performed a study to assess the effect of changing the definition of the operational quantity, personal dose equivalent ($H_p$), in realistic radiation fields in interventional radiology (IR) workplaces. In this type of radiation field, the primary x-ray beam is used to image specific regions of the patient's body that facilitate the diagnosis and the treatment of several conditions. The typical x-ray energy ranges between 60 to 120 kV. The medical staff is exposed to the scatter photons that have a wide range of energies depending on the beam configuration and the patient size.

The study is based on a series of Monte-Carlo simulations of a wide range of X-ray beam parameters that are being used in the clinic. Fluence binned per energy and per direction was calculated with MCNPX using realistic geometry and source spectra in interventional radiology workplaces. The old quantity, personal dose equivalent $H_p(10)$, was calculated according to ICRU report 57 using the simulated fluence per energy and direction.

The results of the study show that the ratio $H_p(10)$ /$H_p$ ranges between 1.5 to 1.8 for all the simulated cases. Higher ratios are observed during low x-ray tube voltages.

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# A practical method for routine eye lens dosimetry of staff in interventional radiology

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## Aims

The largest group of hospital staff receiving the highest doses is found in interventional radiology and cardiology. Transposition of the European legislation into the Belgian legislation in 2020 has lowered the annual eye lens dose limit for professionals from 150 mSv to 20 mSv. Before 2020 the eye lens doses were not routinely measured.

This study uses real-life data instead of phantom measurements to assess the need for routine eye lens dose monitoring and which method is preferred to measure the eye lens dose in daily practice. We also investigated whether the eye lens dose can be estimated adequately by the Hp(10) measurement with the whole body dosimeter (positioned outside the lead apron) or if dedicated eye lens dosimetry should be applied.

## Materials and methods

For 3 cycles of 4 weeks 7 nurses and 9 physicians wore two eye lens dosimeters and a whole body Hp(10) personal dosimeter above the lead apron. The eye lens dosimeters contained DXT-100 (Thermo Scientific) TLDs, placed in either a headset or a headband, and were calibrated to measure Hp(3). All dosimeters were read out in the accredited dosimetry service of UZ Leuven using a Harshaw 6600 Plus TLD reader. After 4 weeks all participants were questioned (Qualtrics MX) about use of personal and collective protection equipment for the eye lens as well as the comfort of the dosimeter holders. Statistics were performed with GraphPad Prism 9.

## Results and discussion

88 eye lens and 46 whole body doses were collected. Headset eye doses (median M=0,58 mSv IQR=1,1-0,17 for physicians and M=0,32 mSv IQR=0,93-0,11 for nurses) were not significantly different from headband eye doses (M=0,51 mSv IQR=1,3-0,13 for physicians and M=0,45 mSv IQR=0,84-0,06 for nurses). Spearman correlation factors between eye lens doses and whole body doses were significant for both dosimeter holders. 2 of 15 participants' estimated yearly eye lens doses would surpass 20 mSv, not accounting for the shielding effect of protection by lead glasses.

The survey showed that the headband (6,69 VAS score) is more comfortable than the headset (6,08 VAS score). The use of lead suspended ceiling screens and lead glasses is not systematically done. These radiation protection measures should be strongly recommended towards the workers.

## Conclusion

Taking the shielding of lead glasses into account, estimated annual eye lens doses of 2 participants surpassed 15 mSv, prompting the need for routine eye lens dose monitoring and the obligation of wearing leaded glasses. The choice between a headband or a headset may be given to the operator. For other participants the Hp(10)/Hp(3) ratio of 0,48 (CI 0,41-0,55) can be used to estimate eye lens doses if this ratio stays relatively constant below 1. Periodic follow-up of this methodology with dedicated eye lens dosimetry is advised.

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# Evaluation of different types of LiF thermoluminescent detectors for ring dosimetry in nuclear medicine

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Nuclear medicine staff is primarily exposed to their hands during radionuclide manipulation. Hence, they often wear ring dosimeters for following up the extremity dose limit. Most ring dosimeters use lithium fluoride (LiF) based thermoluminescent detectors (TLD). Several LiF TLD types exist, each with advantages and disadvantages, depending on the radionuclides handled. The goal of this work was to evaluate the performance of different LiF TLD types for ring dosimetry during manipulation of $^{177}$Lu, $^{68}$Ga and $^{131}$I, which are being used increasingly in theranostics.

Three types of 0.9 mm thick LiF TLDs from RADCARD were evaluated: $^{n}$LiF doped with Mg and Ti (MTS-N), $^{n}$LiF doped with Mg, Cu and P (MCP-N) and $^{n}$LiF with the same dopants in a shallow 30-µm thin sensitive layer (MCP-Ns). Three TLDs of each type were placed in rings attached to 2 cm diameter PMMA rods placed symmetrically around an unshielded polycarbonate syringe. The TLDs were directly facing the syringe at 4.5 cm distance and were exposed to radiation, respectively, from $^{177}$Lu (photons < 200 keV, electrons < 500 keV), $^{68}$Ga (positrons < 1.9 MeV, 511 keV annihilation photons) or $^{131}$I (360 keV photons, electrons < 600 keV) in the syringe. Exposure times and activities were selected to obtain TLD doses between 1 and 600 mGy, well within their linear range. TLD calibration was performed by 50 mGy $^{137}$Cs irradiation at the LNK secondary standard dosimetry laboratory.

This setup was also simulated with the PENELOPE Monte Carlo radiation transport code to obtain absorbed dose reference values, dose depth profiles and energy spectra of the different radiation types hitting the TLDs. The depth profiles allowed to calculate a correction coefficient for light attenuation in the TLD and to assess the dose in the 30-µm thin sensitive layer of MCP-Ns. The energy spectra allowed to calculate a correction coefficient for the relative luminescence efficiency energy dependence. With all these data the theoretical TLD responses could be calculated and compared with the experimental values.

Good agreement within 17% was found between measured and simulated TLD responses for all TLD types and radionuclides. Based on the predicted responses, recommendations could be made for monitoring each radionuclide. For $^{177}$Lu MTS-N is advised because the exposure is dominated by low energy gammas for which MCP-N and MCP-Ns have about 20% under-response. For $^{68}$Ga the exposure is dominated by high energy positrons due to lack of shielding, leading to a significant dose decrease over the TLD thickness. This causes a 20% under-response in MTS-N and MCP-N. Therefore, it is recommended to use MCP-Ns. For $^{131}$I, the selection depends on the shielding. For less than 1-mm plastic equivalent (syringe thickness), low energy electrons form an important contribution leading to a strong dose decrease over the TLD thickness and a 40% under-response for MTS-N and MCP-N. Therefore, MCP-Ns is advised below 1-mm plastic equivalent shielding and MCP-N above because of the higher sensitivity.

# Eye lens dose monitoring in Interventional Radiology: a multi-centre study on endovascular, cardiology and neuroradiology interventional procedures

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Purpose
The eye lens dose limit decrease from 150 to 20 mSv/year for professional exposure, can be relevant for health-care workers involved in interventional procedures. Surgeon eye-lens dose normalized to KAP, for procedure type, is a useful datum for radiation protection optimization.
In the framework of a research project funded by Italian “National Institute for Insurance against Accidents at Work” (INAIL), the eye lens dose has been monitored for most exposed staff members in several endovascular, cardiology and neuroradiology interventional and diagnostic procedure.

Materials and methods
Dose assessment was performed providing three dosemeters to the staff members: dosimeters were attached near to the most exposed eye (LiF:Mg, TLD-100, Ext-Rad, Harshaw Thermofisher), and outside the lead collar and apron (LiF:Mg, TLD-100, Card, Harshaw Thermofisher). These two in order to correlate the eye lens dose to the personal equivalent dose. The first one provides a measure of Hp(3) and the others a measure of Hp(10).
All dosimeters were provided by an ISO-17025 accredited Dosimetry Service.
The investigated procedures are Studies ($KAP_{mean}$ 40 Gy*cm$^2$), Arteriovenous Malformations (AVM, $KAP_{mean}$ 90 Gy*cm$^2$) and Aneurysms (ANE, $KAP_{mean}$ 120 Gy*cm$^2$) for neuroradiology procedures, Coronary and Angioplasty (CA, $KAP_{mean}$ 80 Gy*cm$^2$), Pacemaker implantations and Implantable Cardioverter-Defibrillator (PM-ICD, $KAP_{mean}$ 15 Gy*cm$^2$) for interventional cardiology, EndoVascular Aortic Reconstruction (EVAR, $KAP_{mean}$ 210 Gy*cm$^2$) and the Fenestrated EndoVascular Aortic Reconstruction (FEVAR, $KAP_{mean}$ 360 Gy*cm$^2$) for interventional cardiology.
All procedures were performed with skirt and ceiling-suspended shielding (0.5 mm Pb) except for EVAR and FEVAR where only skirt shielding was present.

Results
The most exposed eye lens dose among all surgical team members for single procedure (normalized to KAP, given in μSv/Gy*cm$^2$), rescaled to a value without the ceiling-suspended shielding, is: 24.2 for Studies, 25.3 for AVM, 26.0 for ANE, 34.0 for CA, 161.6 for PM-ICD, 20, 3 for EVAR and 6 for FEVAR.

Conclusions
In Interventional Radiology the 20 mSv/year dose limit for eye lens exposure may be easily exceeded for the staff members without the use of an effective

shielding equipment and/or Personal Protecting Equipment (PPE). Therefore a personal eye lens monitoring should be considered. The results from a multicenter study on multiple interventional procedures can be used to optimize eye lens radiation protection for medical staff, providing reference dose values and variability ranges.

# The physical model for selected nanodosimetric quantities consistent with radiobiology

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Medicine, trying to catch up with a constantly increasing number of diagnosed cancer cases, introduces new techniques of therapy, especially using non-photon sources of ionising radiation. Unfortunately, the biological outcomes for these methods are much more difficult to predict when considering averaged, macroscopic physical quantities such as absorbed dose, LET, RBE, etc. As the initiation of ionising radiation-induced damage is dominated by interactions occurring at the DNA or within its environs, knowing the distribution of such interactions is crucial to properly assess the biological effects of radiation. Since nanodosimetric targets may be considered as a substitute for the biological target, nanodosimetric measurements could be used to predict the radiobiological response of a given cell line to a different radiation quality when the proper model is established.

The first results of the attempt to connect fundamental nanodosimetric concepts with radiobiological parameters derived from published literature on the survival of the human cancer cells irradiated with ion beam will be presented. To obtain a mathematical formula, we have analysed radiobiological data on the selected, well-established cancer cell line [1] and correlated the biological parameters with nanodosimetric characteristics of the radiation used in these experiments. The nanodosimetric quantities were determined retrospectively based on available information about irradiation conditions using Monte Carlo simulation with the Geant4-DNA physics. For now, the model lacks any intermediate steps including chemistry of free radicals, but provides good data consistency. During the presentation, the proposed physical model to the corresponding experimental data and simulation results will be presented. Further research plans and their limitations will be discussed.

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# Using single-cell DNA sequencing as a dosimetric tool- An exploratory study

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**Purpose:** Radiation-associated tumours show mutation signatures in their genome, which distinguish them from radiation naïve tumours [1]. Our goal is to find if we can identify similar signatures, well in advance, before radiation exposed cells transform into a tumour. We are also investigating if our technique can be used as a dosimetric tool to estimate the radiation quality and dose deposited in an individual during the event of an incident or an aerospace mission.

We know that stochastic interactions of radiation introduce damage and mutations that are unique to each individual cell's genome. But conventional genome sequencing [2] methods such as bulk cell sequencing cannot detect such unique mutations. Therefore, we are examining if single-cell DNA sequencing may be used to reveal the mutational effects of ionizing radiation in exposed cells. We expect to see different mutation pattern for high- and low-LET radiation and thus we hope to discern the radiation quality from genomic mutations in cells.

**Methods:** In this work, four identical samples of a human B-lymphoblastoid cell line were irradiated *in vitro* using 6 MV X-rays from a medical linear accelerator. They were exposed to sham irradiation (control), 0.5 Gy, 1.5 Gy and 3 Gy respectively at a common dose rate. Irradiated samples were incubated for 24 hrs, and then DNA was extracted from approximately 500 cells per sample and subsequently subjected to single-cell whole-genome DNA sequencing technology. The well-characterized genome of our B-lymphoblastoid cell line was used to establish the baseline mutations in our control and to identify radiation-induced mutations in the three other samples. Copy number alterations (CNA) were identified and examined in individual sequence data from all four sample groups. Radiation-induced copy number (CN) gains and losses were counted.

**Results:** We observed a dose-dependent increase in the number of CNA in our sample groups, where the number of CN losses increased significantly with radiation dose. We also observed a dose dependence for the size of the chromosomal aberrations.

**Conclusions**: Our findings suggest that single-cell sequencing techniques may be used to directly examine the mutational effects of ionization radiation in human cells. We are presently working on reproducing these results with repeated experiments. If confirmed, we posit that our strategy of examining DNA anomalies will open up new avenues for radiation biodosimetry.

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# Deciphering beam quality descriptors for proton fields: yd and LETd simulations compared to microdosimetric experiments


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The aim of this work is to analyse how beam quality descriptors - lineal energy (y) and linear energy transfer (LET) - are computed in different codes and to compare them to microdosimetric measurements. We investigate proton beams and use measurements taken with microdosimetric counters as reference. These empirically obtained microdosimetric spectra are compared to simulations using different Monte Carlo particle transport codes. To account for the broadening of the energy spectrum we need to consider the impact of the averaging method: dose- or track- weighting, and the possible inclusion of secondary particles. These effects have a considerable effect on radiobiological modelling (i.e. RBE) based on $LET_d$ (empirical models) or $y_d$ quantities (e.g. Microdosimetric-Kinetic Model) as shown in [3,4].
Monte Carlo particle transport codes capable of simulating microdosimetric spectra and scoring linear energy transfer were used: TOPAS v3.7, Geant4 10.6, SHIELD-HIT12A v0.9.2 and PHITS v3.24. Detailed simulations with high spatial resolution, full physics settings and sufficiently large statistics were prepared to obtain the full energy spectrum. Experimental data were obtained by measurements performed at 62 MeV and 148 MeV at the CATANA and TIFPA beam line, respectively [1,2]. Dose mean lineal energy was derived from spectra measured using a mini-TEPC in pristine and spread-out Bragg-peaks at different depths.
The averaged values of measured $y_d$ for protons are compatible with $y_d$ and $LET_d$ for particle transport simulations. The broad difference in $y_d$ and $LET_d$ by different codes is emphasized, supporting what also was concluded in [3,4]. The inclusion of alpha particles and recoil ions introduces a discrepancy up to 300 % between MC transport codes and measurements. Today, microdosimetric equipment can measure linear energies of 0.1 - 1000 keV/um, which enables comparison between simulations and experiments. Simulation of the particle spectrum provides a way to discuss under what condition $y_d$ and $LET_d$ yield similar values.

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# The Neutron-Induced Carcinogenic Effects Research Program

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Patients undergoing high-energy (>10 MeV) radiation therapy and astronauts in deep space are two population groups for whom whole-body neutron exposure cannot be controlled. As a result, these populations are at risk for neutron-induced carcinogenesis. With this in mind, our group (kildealab.com) has built a research program that we have called Neutron-Induced Carcinogenic Effects (NICE) to: (a) better measure neutron spectra in high-dose-rate radiation therapy environments using multiple neutron detection techniques and (b) better understand the energy-dependence of neutron relative biological effectiveness (RBE) through Monte Carlo (MC) modelling and radiobiological experiments.

**Progress to date:** (a) Using the Nested Neutron Spectrometer™ (NNS, Detec Inc, Gatineau, Canada) incorporating an (active) He-3 detector in current mode, we have demonstrated that it is possible to quickly (≤ 1 hour) measure the in-air neutron spectrum in a radiotherapy bunker [1]. We have independently calibrated the NNS in the high-dose-rate radiotherapy environment by replacing the He-3 detector with (passive) gold foils [2]. Using our NNS setup, we have determined the neutron-production consequences of the flattening filter in high-energy clinical photon beams [3] and we have measured the influence of various beam-shaping parameters on the production of neutrons from clinical electron beams [4]. Presently, we are using track etch detectors (TASL Ltd., Bristol, UK) to examine in-cell-medium and in-phantom neutron spectra with a goal of enabling accurate neutron spectral measurements within samples of human cells irradiated in-vitro.

(b) Using a combination of MC modelling, cell irradiations, and post-irradiation single-cell whole-genome sequencing (ScWGS), we are attempting to better understand the biophysical basis of the energy-dependence of neutron RBE by comparing simulations with the results of real-world cell-irradiation experiments. To date, we have modelled neutron RBE for the microdosimetric endpoint dose-mean lineal energy at various depths in the ICRU sphere [5], akin to the work of the ANDANTE group, and we have modelled neutron RBE for the direct and indirect induction of both complex double-strand break (DSB) clusters and non-DSB clusters using a geometric model of human DNA [6]. Our MC work has involved the use of the Geant4, Geant4-DNA, TOPAS and TOPAS-nBio frameworks and our complete TOPAS-nBio application has been released under an open-source license [6]. We have conducted photon irradiations of human b-lymphoblastoid cells in-vitro with a goal to examine the size and frequency of radiation-induced mutations using ScWGS. Preliminary results (F. Mathew et al., submitted to this conference) show an increase in mutations as a function of dose. Repeat experiments are underway to establish a protocol that will be used to examine neutron-induced mutations.

**Conclusion:** We have established a program to accurately and quickly measure neutron spectra in radiotherapy and to better understand neutron RBE by comparing MC simulations with the spectra of radiation-induced mutations using ScWGS.

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# The Generalized Stochastic Microdosimetric Model, GSM$^2$: a full spatio-temporal stochastic description of particle beams to predict biological effects in a broad dose rate range.

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Presently, the only two radiobiological models used in clinical applications of particle therapy are the Microdosimetric Kinetic Model (MKM) and the Local Effect Model (LEM).
The main limitation shared by both models is the assumption that all physical and biological variables follow a Poisson distribution. This assumption neglects stochastic fluctuations of energy deposition both from cell to cell and from dose fractionation, which can be significant especially in highly mixed radiation fields.
Although some generalizations to overcome the Poissonian assumption have been developed a comprehensive stochastic description of the radiation-induced DNA damage formation and dynamics accounting for both spatial and temporal features of dose deposition is still missing.
To overcome this limitation, we developed a stochastic microdosimetry-based kinetic model (GSM$^2$, generalized stochastic microdosimetry model [1]). Modeling the probability distribution of DNA damages, GSM$^2$ provides a general probabilistic framework to describe the damage formation and evolution.
One of the most relevant strengths of GSM$^2$ is the capability to efficiently treat the several levels of spatio-temporal stochasticity occurring during a protracted irradiation. We extended GSM$^2$ , to consider multiscale processes, coupling the DNA damage evolution with fast reaction kinetics, accounting for radical formations, oxygen consumption and re-oxygenation happening during the delivery of a dose within a defined time structure. The resulting multiscale GSM$^2$ describes the coupled evolution of the system composed by DNA damage and fast Reactive Oxygen Species (ROS) in a broad range of different temporal structures of the dose delivery.
In this way, GSM$^2$ provides an efficient description of acute, split or protracted irradiations that find a natural application in the field of ultra-high dose rate regimes.
An extensive study of the cell survival probability for acute irradiation conditions will be also discussed [2,3], showing how GSM$^2$ encompasses the standard linear-quadratic model, and in addition provides a better ground for mechanistic interpretation. As a relevant consequence, we will show how GSM$^2$ provides a generalization to the multi-hit model, that accounts for non-Poissonian effects and damage repair.

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# Nanodosimetric characteristic of carbon ion beam – experiments and Monte Carlo simulations

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The enhanced biological effectiveness of carbon ions most pronounced in the Bragg peak is related to the ionisation pattern of the carbon ion track in the nanometer scale. In view of the impending radiation therapy with carbon ions, the stochastic nature of the track structure of this type of radiation is of particular interest. To this end, experimental nanodosimetry has been applied and is still being developed. Its techniques allow for investigation and modelling of the interaction of ionising radiation at a scale comparable to the size of subcellular structures like DNA.

Experimental studies of a nanodosimetric characteristics of carbon ions were conducted during the last years using the Jet Counter (JC) nanodosimeter developed at the National Centre of Nuclear Research, Poland [1, 2]. The series of experiments with the carbon ion beam delivered with the cyclotron at the Heavy Ion Laboratory, University of Warsaw allowed for the improvement of JC detection system. The experimental setup upgrades resulted in extended range of target sizes irradiated with carbon ions of energies in the range 12-80 MeV. While only of part of the collected data has been published so far [3, 4], a comprehensive overview of the results of all experiments with carbon ion beam has been prepared and it will be presented along with a comparison with Monte Carlo simulations. A detailed description of an experimental setup and solutions will be provided.

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# Calibrations and irradiations in terms of new ICRU operational quantities for radiation protection possible in photon and beta reference radiation fields

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**Keywords:** New operational quantities, calibrations, irradiations, photons, betas

The International Commission on Radiation Units and Measurements (ICRU) has recommended new operational quantities for radiation protection measurements in its report No. 95 [1]. However, these have not been officially adopted yet at international (IAEA, International Atomic Energy Agency) or European (EU, European Union) level by a corresponding amendment to the respective radiation protection basic standards. To be able to estimate the possible consequences of such an adoption of the proposed quantities metrologically at an early stage, the values required for this were calculated. With them, calibrations and / or irradiations of radiation protection dosemeters in the newly proposed radiation protection quantities with photon and/or beta radiation are possible [2],[3].

The calculated values show that a change of the measurands, depending on the type of radiation and energy range, will have a massive impact on the response of previously ideal dosemeters:

- This is true for photon radiation for dosemeters estimating dose in local skin and the dose in the lens of the eye in the rather irrelevant energy range below 10 keV and
- especially for dosemeters estimating effective dose in the important middle energy range up to about 70 keV.
- For beta radiation, massive changes will occur in the response for dosemeters for the estimation of the dose in the lens of the eye.

Therefore, a change of the official quantities from the current to the new ones, proposed by the ICRU, should only be considered after carefully balancing the advantages against the disadvantages. A more detailed discussion has already been started [4].

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# EURADOS Project on the Impact of the Proposed ICRU Operational Dose Quantities

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The International Commissions on Radiation Units and Measurements (ICRU) and on Radiological Protection (ICRP) have recently published a joint report [1] recommending new operational quantities for use in the system of radiological protection. The new operational quantities have been devised to address certain problems with the existing ones, including the need to cover a wider range of radiation types and energies, for example arising from the increasing use of proton therapy in clinical procedures. Also related to changing practices is the increased importance of doses at diagnostic x-ray energies (several tens of keV), where the reduced dose limit for the lens of the eye [2,3] and the more frequent use of interventional procedures make the over-estimates of dose given by the existing quantities less acceptable. The new operational quantities are conceptually different from the existing ones, being defined using the same anthropomorphic voxel phantoms as are used to derive the protection quantities [4]. ICRP have carried out a consultation process and ICRU have revised the report in the light of comments received.

As part of its strategic research agenda the European Radiation Dosimetry Group, EURADOS (www.eurados.org), seeks to contribute to the development and understanding of fundamental dose concepts, such as the topic of operational quantities. Accordingly, EURADOS is carrying out a project to evaluate the impact of the proposed ICRU operational quantities and to make recommendations for their application. The planned report will analyze the differences between the new and existing quantities before going on to examine impact and application in the areas of: radiation protection practice, dosemeter and instrument design, calibration and reference fields, European and national regulation, and current published standards.

The new quantities will achieve the benefits of wider radiation type and energy coverage, and of improving representativeness in the diagnostic/ interventional photon energy range below about 80 keV. The biggest negative impact will be in the area of dosemeter and instrument design. The changes needed to achieve good responses to the new operational quantities will range from simple re-calibration to radical re-design; and some types of dosemeter may become obsolete. Investments are therefore required to achieve the aforementioned improvement in the system of metrics. The present report explores this and other impacts.

We support the recommendation that the introduction of the new quantities should be phased over tens of years. Not only will this provide time for the costs and benefits to be fully assessed and the necessary research to be carried out, it will also allow for consideration of the parallel development of the planned new recommendations from ICRP.

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# Conversion coefficients for $H_p$(10, ROT) and $H_p$(10, ISO) and the implications for ICRU Report 95

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ICRU Report 95 (ICRU, 2020) proposed significant changes to the operational quantities that are jointly published with ICRP and explicitly approved by them (Clement et al., 2021). They are hence likely to be included in the updated ICRP recommendations and consequently find their way into the new Basic Safety Standards produced by the IAEA and EURATOM. These new operational quantities will have implications for many areas of radiation protection, but the changes to the quantities for personal dosimetry will be of particular significance.

The new quantity *personal dose*, the replacement for *personal dose equivalent* or $H_p$(10), is defined in the reference phantoms and based on a complex averaging process. The published data include the following fields: 0°, avg(±15°), avg(±30°), avg(±45°), avg(±60°), avg(±75°), avg(±90°), 180°, ROT and ISO. The last four of these fields cannot be directly compared to $H_p$(10) because there are limited published data to compare with since they were not included in ICRU Report 57 (ICRU, 1998) or ICRP Publication 74 (ICRP, 1996).

Neutrons in the workplace exhibit more scatter than photons, which changes their direction distribution considerably, but ROT fields are relevant to both due to the movement of workers. To help understand the implications, new calculations of $H_p$(10) have been performed to enable direct comparisons with ICRU95 data. The impacts are discussed in terms of the UKHSA personal dosemeters (Hager et al., 2017, Eakins et al., 2008) using calibration data and workplace field measurements. The complex question of whether the new quantity will require more than one dosemeter for workers in highly scattered workplace fields is discussed.

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# Dosimeter Angular Response and the ICRU 95 Quantity Personal Dose

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In Report 95,[1] the International Commission on Radiation Units and Measurements (ICRU) recommended the new unit of personal dose which is based on the angular-dependent values of the effective dose. The values in that report are the same as those reported by A. Endo [2] for angles from 0 to $\pm$90 degrees and at 180 degrees. Values of the angular response of track-etch and TLD-based dosimeters mounted on slab phantoms were previously measured in several calibration neutron fields (Unmoderated $^{252}$Cf, $D_2O$-moderated $^{252}$Cf, polyethylene-moderated $^{252}$Cf, Cadmium-covered polyethylene moderated $^{252}$Cf, PuF and Pu-Be) by the authors. The relationship between these measurements and the recommended personal dose equivalent quantities as a function of angle are compared. The potential for using existing dosemeters to replicate the angular response of the newly recommended quantity personal dose equivalent will be discussed based on these comparisons.

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# Investigation of Cf-252 Sources with a Ge Gamma Ray Detector

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Radiation protection is one of the crucial issues concerning the disposal of spent fuel casks in dedicated repositories. Several devices such as neutron moderation spheres and dosimeters using thermoluminescence detectors (TLDs) or optically stimulated luminescence (OSL) detectors are commonly introduced in dose monitoring. However, those devices show relatively large uncertainties. To improve the assessment of radiation fields and related dosimetry, Germanium (Ge) gamma ray detectors are considered. In comparison to TLD and OSL measurements, Ge detectors provide high-resolution gamma spectrometry, i.e. the analysis of the pulse height spectra increases the accuracy determination of the gamma ray quantities.
The aim of this work was to investigate the mixed neutron-gamma field of Cf-252 sources by employing a Ge gamma ray detector. To determine the properties of Cf-252 sources and in particular, to assess whether the neutron emission yields can be properly determined.
Previously, gamma-ray spectra of Cf-252 and the information they contain have been studied, but with the focus on aged sources [1].
In our investigations, experiments were performed with Cf-252 sources, including a young source, and an available high-purity Ge detector. The Nucleonica Nuclear Science Portal [2] was employed. Tools of “Nucleonica” such as e.g. “Decay Engine”, “Gamma Spectrum Generator” allow for calculating the decay products of Cf-252, their age dependent activities, as well as gamma-ray spectra with a detector response model. The inclusion of simulated gamma-ray spectra offers a new advantageous approach.
New encouraging results concerning young and very old sources are presented.

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# Calibration of a large-size wide-range neutron spectrometer for environmental application

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Over the last year, Politecnico di Milano has developed a wide-range spectrometer for cosmic neutrons which can be employed for soil moisture investigation [1]. The instrument is based on the thermal neutron counter M800 manufactured by Arktis, which detects thermal neutrons exploiting the capture reaction of $^{6}Li$ and the scintillation mechanism of $^{4}He$. The M800 is a Ø14×80cm cylinder divided into 8 sectors which can be read independently. The sectors were coupled in pairs, obtaining four regions, called thermal, epithermal, fast, and high energy. Each one of them is covered by a different moderating material to enhance its sensitivity over the corresponding region of the energy spectrum. Since the detection of neutrons above 20 MeV requires the employment of nuclear reactions occurring on heavy metals like lead, the detector, despite being dismountable, has a considerable weight (around 100 kg). The response functions of each region were calculated via the MCNP6.2 code and are given as an input to an unfolding code written in Python which calculates the energy spectrum employing the GRAVEL algorithm.
This paper discusses the calibration of the detector in two reference fields:

1. Monoenergetic neutrons (from 74 keV to 14.8 Mev) at PIAF facility at PTB.
2. Neutron spectrum resembling the cosmic one generated at the CERF facility at CERN.

For monoenergetic fields, the detector was held in the measuring position with the support of a crane and placed at 3.7m from the target generating the neutrons. The calculated spectra and ambient dose equivalent H*(10) were compared with the reference values, finding an overall good agreement (the maximum error on the H*(10) was around 10%). Still, some other aspects were noticed: in the most energetic fields (2.5 Mev, 14.8 MeV) the thermal end epithermal regions recorded many more counts than expected by the response functions. This aspect is not deeply penalizing but requires further investigation. We believe that elastic scattering between neutrons and helium might cause scintillation in $^{4}He$. This contribution to the detection efficiency is not considered by the current version of the response functions. During the irradiation at the CERF facility, the detector was placed on a wooden trolley in one of the reference positions of the side shielding. The size of the detector did not allow a perfect alignment and the instrument was shifted towards the floor. The calculated spectrum shows an evident overestimation of the epithermal component of the flux, and the measured H*(10) tends to underestimate roughly of the 10% the reference value. The discrepancy might be explained by the difference in positioning: the detector has a bigger size with respect to the reference position and is expected to be more affected by the neutrons scattered on the floor. In addition, the neutron field may be inhomogeneous over such a wide volume.

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# ANALYSIS OF THE EURADOS NEUTRON INTERCOMPARISON RESULTS ACCORDING TO RECENT ISO STANDARDS

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The European Radiation Dosimetry Group, EURADOS, has carried out several intercomparison exercises in the past that qualify as proficiency tests for different dosimetry systems and types of radiation. The first neutron dosemeter intercomparison was held in 2012 (IC2012n) and was followed by a second one in 2017 (IC2017n). 31 Individual Monitoring Services (IMSs) entered 34 dosimetry systems in IC2012n and 32 IMSs entered 33 dosimetry systems for IC2017n. Such exercises provided a rare opportunity to see how neutron dosemeters perform [1]. For the IC2012n exercise, there were no applicable performance standards for neutron personal dosemeters inter-comparison. The need for such standards was emphasized by the IC2012n organization group in their conclusions [2]. The conclusions were at that time driven only on a “factor-2” criterion, set internally the EURADOS Organization group, based on ICRP statement on allowance of wider uncertainty for neutron dosimetry as compared to photon dosimetry. Fortunately, the ISO 14146 standard was updated in 2018 [3], now including neutron dosimetry in the performance criteria for a performance test. Whilst IC2017n exercise could be analyzed in accordance with the requirements given by this new document, it is of interest to re-analyze the results of IC2012n to quantify any modifications of the conclusions. In parallel, ISO standard 21909 was also largely modified since 2012 and such deep revision was published in 2015. This document gives performance requirements for type testing of passive neutron dosemeters [4]. Although the philosophy of the tests is very different to the ISO 14146 standard, the revised 21909 highlighted similar issues for neutron dosimetry to those raised by ICn exercises such as the need for some IMSs to have information on the workplace fields, and the importance of testing dosemeters at low dose levels.

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# Quality assurance for dosimetry system based on Instadose+ and Instadose2 dosemeters: long-term stability of the dosemeters and results of international intercomparisons

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Mirion Dosimetry Services offers dosimetry services based on Instadose dosemeters to more than 300.000 wearers world-wide. Instadose dosemeters are equipped with Direct Ion storage (DIS) detectors and communicate the dose readings via internet to the dosimetry service. Since 2018 the European branch of Mirion Dosimetry Services holds an ISO 17025 accreditation for providing dosimetry services based on the connectorless Instadose+ and Instadose2 dosemeters.
To guarantee the dose results are being reported within stated uncertainties a variety of quality assurance procedures have been put in place, such as daily read analysis. This paper describes two important quality assurance pillars: methods and results of tests of the long-term stability of the dosemeters, and the results of international intercomparisons.
For testing the long-term stability of the Instadose dosemeters different tests are being made. In the first type of test a set of dosemeters is being irradiated quarterly using an in-house available calibration facility. Another test consists of recalibration of dosemeters which have been in service at customers for different periods of time, up to several years. These dosemeters are being returned to the service e.g. because of cancellation of the service by the customer. The results of Mirion Dosimetry Services in Europe are being supplemented by data of almost 10 years of operation of the Instadose1 dosemeter by Mirion Dosimetry Services in USA.
The results of these tests show excellent long-term stability, deviations being smaller than about 5%, even after periods of 5 years. This underpins the recalibration period of 5 years which is being used in the routine service.
Mirion Dosimetry Services participated in the international intercomparisons for whole body photon dosemeters organised by EURADOS in 2016, 2018 and 2020 with both Instadose+ and Instadose2. All results fall within the performance criteria set by ISO 14146, and show good reproducibility and linearity up to dose values of about 500 mSv.

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# EURADOS intercomparison IC2020$_{ph}$ on whole body dosemeters for photons

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The European Dosimetry Group (EURADOS) organized international intercomparisons for whole body dosemeters on a regular basis since 2008. The most recent intercomparison was announced for the year 2020, however, it was postponed to the end of 2020 and the beginning of 2021 due to the pandemic of COVID-19 restrictions. 112 monitoring services from 50 countries with 132 dosimetry systems participated. All dosemeter irradiations were performed by the accredited calibration laboratory of the Greek Atomic Energy Commission (EEAE). The irradiation plan consisted of nine irradiation setups with five different photon radiation qualities (S-Cs, S-Co, N-150, W-60, W-80) and two different angles of irradiation incidence (0$^o$ and 60$^o$).

The present work describes and analyses the individual results for the personal dose equivalent quantities $H_p(10)$ and, if submitted, $H_p(0.07)$ for all participating systems. More over the difficulties occurred from COVID-19 crisis during intercomparison are discussed.

At the time of the submission of this abstract, the analysis of the results is still going on. The preliminary statistical results show a satisfactory performance with the medians of all $H_p$ (10) and $H_p$ (0.07) response values very close to unity. However, also there are few IMs with significant large variations.

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# An Italian network for dosimetry services

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In Europe there are several entities, either public or private companies, that run a personal dosimetry service to monitor workers exposed to ionizing radiation. X/gamma external monitoring is the most requested one [1]. These entities are often referred to as Individual Monitoring Services (IMS).
European guidelines [2] state that each Member State has to approve IMSs, arranging entities, procedures and rules for that. Guidelines give some recommendations for requirements for approval: the main is the compliance with ISO 17025. Such a compliance is called accreditation. It is also strongly recommended from IAEA [3].
Accreditation requires, as one of the QA tests, to perform routinely intercomparisons in order to have an external test of the whole process. In Europe there is only Eurados, performing international self-sustained intercomparisons every 2 years.
In Italy the body able to approve, the procedures, the rules, are not yet stated. Consequently, no entity yet proponed a national dosimetric intercomparison.
Nevertheless, many IMSs implement quality programs and participate in Eurados intercomparisons on a voluntary basis. Furthermore, three IMSs at now achieved ISO 17025 accreditation, thus needing to perform periodic evaluation.
This work is an attempt to set up a self-sustained Italian intercomparison. The aim is to set up procedures and rules to make it not a spot event but a regularly scheduled activity, involving all accredited services and thus following the ISO guidelines [4, 5]. The first “friendly” Italian intercomparison with the results and the problems rising up is shown.
A second result that is expected is a networking one: when performing intercomparisons participant's meeting are planned, to present and discuss results. Subjects of such meetings are both technical and managerial issues, so a network of Italian IMS, that is actually completely missing due to the lack of binding requirements for the IMSs, is expected from this work.

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# Measurement of the neutron spectrum of 241AmB, 241AmLi and 241AmF sources

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**Abstract:** Measurement of the neutron spectrum was performed using the Bonner multisphere methodology, which consists of using a 6LiI(Eu) thermal neutron detector. Counts from neutron radionuclide sources 241AmB, 241AmLi and 241AmF were performed with six different spherical moderators around the detector, and through these measurements the neutron spectrum was obtained using software developed by the neutron metrology laboratory of the Radiation Protection and Dosimetry Institute–LN/IRD, NeutraLN that uses the technique of neural networks. The importance of characterizing the energy spectrum of the aforementioned sources is essential for expanding the LN/IRD measurement capacity, although the emission intensities of these sources are small when compared to other sources used for use in device irradiation and calibration such as those often used 241AmBe and 252Cf. The mean conversion fluence to equivalent dose of those sources are also presented.

# The *EURADOS CR-39 Quality* task for the optimization and harmonization of personal neutron dosimetry with *CR-39*


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The most recent intercomparisons for neutron dosimeters organized by *EURADOS* (European Radiation Dosimetry Group) [1][2] show that *CR-39* nuclear track detectors are among the most widespread devices used for personal neutron dosimetry. In fact, they allow to obtain simple and cheap passive dosimeters, suitable for individual monitoring in a wide range of workplaces in different sectors, from nuclear industry to medicine or research. However, one main issue that limits the performances of *CR-39*-based dosimeters is the variable material quality of the *CR-39* detectors.

In 2020, the *EURADOS* Working Group 2 (*WG2*) launched the *CR-39 Quality* task: a new project aimed at improving and harmonizing personal neutron dosimetry with *CR-39* in Europe. The action is in line with the general goal of enhancing personal neutron dosimetry techniques, as stated in the latest version of its Strategic Research Agenda (*SRA*) [3]. The new task gathers experts from six institutions located in five European countries. Furthermore, it includes a collaboration with some of the most important dosimeter grade *CR-39* manufacturers worldwide, which facilitates the direct dialogue between producer and consumer. The research activities related to the *CR-39 Quality* task are carried out jointly by the institutions participating in the project exploiting sample material supplied by the manufacturers collaborating in the task. The project envisages several studies and experimental campaigns to investigate and compare the currently available *CR-39* materials, to relate the physical and dosimetric properties of a given material to its composition and manufacturing process, and to seek for improvements in the production process and quality assessment tests to reach a more consistent material quality, thus obtaining an optimized *CR-39* for personal neutron dosimetry applications. In parallel with these activities, the task also intends to individuate and disseminate the best practices linked to the use of *CR-39* as a neutron dosimeter.

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# Eurados intercomparisons on whole-body dosemeters for photons from 2008 to 2020: Analysis and comparison of general results

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Starting in 2008 the European Dosimetry Group (EURADOS) has been performing international intercomparisons on photon whole-body dosemeters for individual monitoring services [1] [2]. These intercomparisons were organised (on a biannual basis) starting 2008 up to 2020 now, each time with a similar set-up but with small alterations in the subsequent irradiation plans. The response for $H_p(10)$ and $H_p(0.07)$ was tested for different radiation qualities, angles of incidences (0º - 60º) within a wide range of dose values (0.5 mSv – 500 mSv), thus energy and angular response, linearity, reproducibility and mixed-field response were checked.

Within the last 12 years almost 15000 whole body dosemeters were irradiated and the corresponding response values evaluated. With an increasing number of participants and participating systems (from 62 to 132!), this intercomparison action has become an important tool for individual monitoring services to test their whole-body dosimetry systems, compare their results with other services or systems and to improve the quality of their dosimetry.

The paper presents and compares the general results of these seven intercomparisons and analyses the dosimetric results for the participating system types.

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# Results and Analysis of Intercomparisons study for individual monitoring services of Latin-American and Caribbean Region

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The International Atomic Energy Agency (IAEA), through the regional Technical Cooperation project RLA9088 (Strengthening the radiation safety infrastructure for Latin-America and the Caribbean) supported eighteen individual monitoring services from fourteen countries of the region to participate in the 2020-Eurados whole body dosemeters intercomparison for photon radiation. The aim of this paper is to present the results obtained and contribute to the technical improvement of personal dosimetry services in the Latin-American and Caribbean region. The dosimeters were irradiated at the Ionizing Radiation Calibration Laboratory of Greek Atomic Energy Commission with gamma radiation from S-Co and S-Cs, and with X-ray radiation qualities W-60, W-80 and N-150. All irradiations were performed according to the international standard on the appropriate ISO water slab phantom (30x30x15 cm$^3$), at distances of 100-300cm from the source, depending on required Kerma air rate, and using a Buildup PMMA plate of 0.3x30x30 cm$^3$. The reference point of the personal dosimeters was the frontal surface of the slab phantom. The ratio (R) between the reported dose values from the laboratories and the reference doses for Hp(10) were calculated. All services participating in this study use TLD systems except one that uses OSL dosimeters. The results were analyzed according to the performance criteria established in the standard ISO 14146:2018 [1], commonly known as “trumpet curves”. Results exceeding these performance limits were considered outliers.
Data analysis shows that 93.4% of the results are within the acceptance range and therefore 6.6% are inappropriate. This percentage is like that found by the intercomparison carried out by EURADOS in 2018 for whole body dosimetry, where 6% of the results obtained with the TL dosimetry systems were atypical [2].
The evaluation of the trumpet curves shows that only three laboratories presented few dosimeters results outside the limits of acceptability, indicating that 81.3% of the evaluated laboratories with the personal dosimetry system for Hp (10) are in accordance with the requirements of ISO 14146:2018. The participation in this intercomparison EURADOS exercise gave the Latin American and Caribbean laboratories the opportunity to show compliance with their own quality management system, compare results with other participants and develop plans for improving their dosimetry systems. It also give the assurance that occupational doses are being measured properly and following the international standards in the region.

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# Past, presents and future of thermoluminescent dosimetry in Poland – a review article

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The beginnings of thermoluminescence dosimetry date back to Maria Skłodowska-Curie observing the glow of natural calcium fluoride during heating and early exposure to ionizing radiation from a radium source [1]. The 1950s, with Farrington Daniels, saw the first application for dose measurement and dose levels from various types of ionizing radiation [2]. In the 1960s, the development center of thermoluminescence dosimetry in Poland was the Dosimetry Laboratory of the Institute of Nuclear Physics leaded by Tadeusz Niewiadomski [3]. In the following years, the original LiF: Mg, Ti (MTS-N) detector was developed, and then the highly sensitive LiF: Mg, Cu, P (MCP-N). Currently, all dosimetry services in Poland use both types of detectors on a mass scale in individual, environmental and clinical dosimetry. The paper will also present the non-standard applications of thermoluminescence in determining the elapsed exposure time, second readout of doses, planar 2D readouts and the development of on the eye lens dosimetry. The presented work will be a review, it will contain the achievements of the 50-year history of thermoluminescence dosimetry in Poland as well as the present and future development elements.

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# Individual monitioring with BeOSL dosemeters: state of the art and future developments

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Mirion Technologies DSD is using passive dosemeters based on BeOSL technology for individual monitoring of approximately 600 000 workers in 22 different countries. In this presentation we describe latest additions of a second-generation reader-eraser combination with additional capabilities for measurement of new BeO detector types and improvements to the highly automated technical infrastructure of the BeOSL dosimetry system, distributed under the brand name Dosimetrics and used to process dosemeters at the IMS.
We show the portfolio of BeOSL dosemeters and detectors with recent additions of extremity dosemeters for finger ring and eye lens dosimetry, including prototypes for measurement of beta radiation and research activities investigating the possibility of neutron measurements with BeO detectors using neutron converters.
Furthermore, we report on challenges for the IT systems of the IMS to accommodate requirements for accreditations and approvals of the dosimetry service, reporting of dose results in different countries, and to provide state of the art data security and traceability in compliance with international regulations.

# Recent Developments at the Calibration Laboratory for Radiation Protection Instruments and Dosimeters at the Paul Scherrer Institute

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The Calibration Laboratory at the Paul Scherrer Institute is responsible for the calibration and verification of radiation protection instruments and dosimeters for its own use and for external customers such as nuclear power plants and hospitals. The Laboratory is authorized by the Federal Institute of Metrology METAS to perform legal verifications and is accredited as an inspection body according to ISO 17020 and as a calibration laboratory according to ISO 17025 by the Swiss Accreditation Service SAS. In addition to routine activities (e.g., calibration of dosimeters, irradiation of passive and active dosimeters on phantoms), the Laboratory is involved in various research and development projects with the goal of supporting the increasingly complex calibration needs and to comply with new regulations, particularly those related to clearance limits in free release measurements. Here we present an overview of these currently on-going projects, which include the investigation of Monte-Carlo methods for the calibration of clearance monitors or the development of new calibration procedures for wipe test counters. We also discuss the challenges encountered in these projects, since different stakeholders, ranging from legal authorities to equipment manufacturers and users, are involved in the calibration process.

# The accreditation of the Politecnico di Milano as a provider for proficiency testing in external radiation dosimetry

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In Italy there are several Individual Monitoring Services (IMS), either public or private companies, performing a personal dosimetry service to monitor workers exposed to ionizing radiation: the external monitoring is the most requested one.
According to the guidelines of European Commission [1], an approved dosimetry service is a body responsible for the calibration, reading or interpretation of individual monitoring devices. Their capacity to act in this respect is recognized by the competent authorities.
An approved dosimetry service should implement a quality assurance programme, including also the participation in national or international intercomparison exercises. This is mandatory for IMS, which obtained the EN ISO/IEC 17025 accreditation [2].
In Europe, there is only EURADOS performing international intercomparisons on photon whole-body dosemeters every two years and additional intercomparisons for extremity, eye-lens and passive area dosemeters with different periodicity.
Currently in Italy, there is still no body able to approve a dosimetry service. Nevertheless, many IMSs implement quality programs and some of them achieved EN ISO/IEC 17025 accreditation.
To respond more systematically to the needs of Italian IMSs, the laboratory of Radiation Metrology of Politecnico di Milano has recently obtained the accreditation as proficiency testing provider, according to the standard EN ISO/IEC 17043 [3], from the Italian accreditation body ACCREDIA
The process to obtain the accreditation include the competition of proficiency test to submit for evaluation by the ACCREDIA officers. In this work, the authors present: (i) a summary the pilot proficiency test and its organization to differentiate form the one regularly proposed by EURADOS, (ii) the convenience, for this specific kind of activity, to request the accreditation of flexible scopes [4], (iii) the lesson learnt during the whole process leading to the accreditation as proficiency testing provider.

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# ASSESSMENT OF THE IMPACT ON AN OSL DOSIMETRY SYSTEM OF THE NEW OPERATIONAL DOSE QUANTITIES PROPOSED IN THE ICRU 95

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In 2020, a new ICRU report on Operational Quantities for External Radiation Exposure was presented by ICRP. This document presents a new concept for definition of the operational quantities which is radically different to the previous ICRU concept. This change will impact the metrology of occupational dosimetry for the radiation protection of the worker. For instance, the conversion coefficients (personal dose equivalent / fluence) for photons change significantly below 50 keV up to a factor exceeding 5. The work presented consist to assess the impact on an individual dosimetry system for photon exposure based on an OSL technology with $Al_2O_3$:C sensor. This impact is analysed considering the criteria of the EN62387:2016 (which defined the performance and test requirements for personal dosimetry). It will be presented also a retrospective impact on the dose reported on the previous monitoring periods at the French Landauer laboratory.

# Optimization of calibration interval based on equipment metrological history

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The purpose of this presentation is to describe a method for optimizing the calibration period of a dosimetry measurement system. The systematic and analytical method developed provides a justification to the calibration interval based on experimental data and uncertainties. Its main input is the metrological history of the equipment that is then used to predict the drift of the measurement system. The method is used to make the calibration operation more efficient over time in terms of cost and metrology. After a description of the method itself, the example of the OSL dosimeter reading process will be discussed and results will be shown. The method developed can be applied to all measurement processes no matter the technology used.

# Personal online dosimetry using computational methods: the PODIUM Project

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**ABSTRACT**

Individual monitoring of workers exposed to external ionizing radiation is essential to allow application of the ALARA principle and follow up of the official dose limits. However, large uncertainties still exist in personal dosimetry, especially for neutrons and for inhomogeneous fields. Also, many practical problems exist for personal dosimetry, with many dosemeters getting lost and the reluctance of many workers to wear one or more dosemeters.

The objective of the PODIUM project is to improve personal dosimetry by an innovative approach: the development of an online dosimetry application based on computer simulations without the use of physical dosemeters. Operational quantities, protection quantities and radiosensitive organ doses (e.g. eye lens, brain, heart, extremities) will be calculated based on the use of modern technology such as personal tracking devices, flexible individualized phantoms and scanning of geometry set-up. When combined with fast simulation codes, the aim is to perform personal dosimetry in real-time. Parallel to this, a different approach was planned with pre-calculated fluence to dose conversion coefficients for phantoms of different statures and postures.

We applied and validated the methodology for two situations where improvements in dosimetry are urgently needed: neutron workplaces and interventional radiology. An online application in which we calculate individually the level of occupational exposure is developed. For that purpose, the spatio-temporal radiation field, including its energy and angular distribution, needs to be known. We use input from dose monitors in the neutron workplace and radiation dose structured reports (RDSR) from the x-ray machine used in interventional radiology and we capture real movements of exposed workers and transfer this to the calculation application.

This paper will describe the achievements of the PODIUM projects in this new approach for personal dosimetry. We will show the results from the validation and test measurements in different hospitals, and in 2 workplace fields with significant neutron exposure.

The availability of the proposed online personal dosimetry application shall overcome the problems that arise from the use of current passive and active dosemeters. Such

limitations include the uncertainty in assessing neutron and photon doses when part of the body is shielded, the delay in calculating the doses and the situation where workers position dosemeters incorrectly. In addition, it will increase awareness of radiation protection among workers and will improve the application of the ALARA principle.

# Computational personal dosimetry at a realistic neutron workplace field

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Personal dosimetry for radiation workers is of fundamental importance for radiation protection and is currently almost exclusively performed by physical personal dosimeters. These are characterized by several important drawbacks such as limited or no spatial information and imperfect energy and angular dependence of response on top of several practical issues. Some of these drawbacks could be avoided by using 3D cameras for tracking of workers, modelling the workplace and its radiation field, and then computing the worker radiation doses by Monte Carlo radiation transport simulations. The PODIUM (Personal Online DosImetry Using computational Methods) project was setup to demonstrate the feasibility of this innovative dosimetric approach, specifically for interventional radiology and neutron workplace fields.

The goal of this work is to demonstrate the feasibility of the computational dosimetry approach developed within the PODIUM project at a realistic neutron workplace field. A transport container with spent MOX fuel needles in a controlled area at SCK CEN was selected as realistic neutron workplace field for this feasibility study. An MCNP6.2 model of this workplace was developed and successfully validated by comparison of simulations with extensive measurements. After validation the model was used to calculate an effective dose rate map of the workplace field. A single 3D camera was setup at the workplace to track the workers. Using a Python tool, the effective dose rate map and the tracking file from the camera, the effective dose of the tracked worker could then easily be calculated.

This work clearly demonstrated the feasibility of the computational dosimetry approach developed within PODIUM at a realistic neutron workplace field. The most challenging and time-consuming step in this approach was the setup and validation of the simulation model for calculation of the effective dose rate map. This required several weeks of work. Future work will focus on exploring options to speed up the effective dose rate mapping.

# Dose assessment with fast Monte Carlo codes in interventional radiology

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Individual monitoring of radiation workers in interventional radiology is an important issue for radiation protection. The rising number of procedures using these less-invasive techniques makes essential a good quantification of medical staff exposures and the continuous improvement in radiation protection tools. In particular, the European Directive 2013/59/Euratom [1] establishes a new eye lens dose limit of 20 mSv in a year for occupational exposure which is the same that the limit for the effective dose and, thus, is more likely to be approached or even exceeded.

In interventional radiology procedures the radiation field is highly inhomogeneous and therefore obtaining representative dose measurements can be difficult. Computational dosimetry offers an alternative to physical dosemeters for dose assessment in this field. However, in a typical interventional procedure where tens to hundreds of irradiation events are performed, dose calculations using standard Monte Carlo (MC) codes are too long.

To overcome this limitation, two fast MC simulation codes, PENELOPE/penEasyIR and MCGPU-IR, were developed at UPC and validated against the standard PENELOPE MC code and with measurements during a realistic interventional procedure [2].

This study investigates further the performance of these two fast MC programmes. In particular, the influence of the shielding materials when assessing the workers' doses is analysed.

Several dose measurements in realistic set-ups have been carried out in a Secondary Standard Calibration Laboratory and in a real clinical interventional room with a Philips angiography system. Active (Raysafe i3, Thermo-Electron EPD Mk2.3 and Mirion DMC3000) and passive thermoluminescence personal dosimeters have been used to compare experimental measurements and calculated data. Irradiations with shielding devices mimicking ceiling suspended shields have been carried out in the calibration facility in order to analyze the capability of the computational tools to reproduce the experimental values when this type of protective equipment is used.

First results show differences in terms of $H_p(10)$ about ± 30 % between experiments and calculation and variations up to 68% between different physical dosemeters when measurements are performed behind shielding.

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# Obtaining the national metrological traceability chain associated to the dosimetry of the eye lens by creating high-precision dosimetry phantoms

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As is known, one of the most radiation-sensitive parts of the human eye is the eye lens, which could gradually show degradation (e.g., radio-induced cataract) by exposure to ionizing radiation. To provide support for monitoring the absorbed dose by the eye lens, the International Organization for Standardization has issued ISO 15382:2015 [1]. To this end, knowing the precise magnitude of the absorbed dose by the eye lens is of great importance. Furthermore, the applied dosimeters have to be appropriately calibrated and traceable to the national and international dosimetry standards. At a national level in Romania, *"Horia Hulubei National Institute for R&D in Physics and Nuclear Engineering (IFIN-HH)"* is in charge of performing all the ionizing radiation-related metrology activities.

In this work, an innovative metrological method based on using modern 3D printing techniques to create high-precision human head dosimetry has been proposed. In the first phase of this project, a human head phantom has been created by a 3D printer, with special attention to the 3D eyeball, having the same geometry and composition as the real eye. Five standard dosimeters (BeOSL) were attached to the forehead, both eyes, and both temples of the head phantom. Then, it was exposed to a standard ionizing radiation field (Co-60 radionuclide, certified source) from 1.5 meters distance in three sets of experiments for 115, 230, and 345 minutes, respectively. The horizontal axis of the source was aligned with the forehead dosimeter. After each exposure, the $H_p(0.07)$ and $H_p(10)$ quantities absorbed by each of those 5 dosimeters were determined and registered. Then, the same experimental setup including source, collimators, phantom, and dosimeters was modeled by Monte Carlo technique and the 3 experiments were simulated. This has been repeated both experimentally and computationally, while the horizontal axis of the source was aligned with the dosimeter attached to the right temple of the head phantom. The experimental and computational results (the absorbed dose value by each dosimeter) were compared, considering all the involved parameters, and the observed discrepancies were interpreted.

The second phase of this project (ongoing) is related to the determination of the uniformity of the ionizing radiation field, meaning that a special pair of glasses with passive detectors will be worn by the head phantom. After gradually reading the dose values along to the radiosensitive parts of the glasses, the uniformity of the radiation field will be determined. This field uniformity will be double-checked as well by using the Monte Carlo method. The final step of the experiment is providing traceable, reliable and precise calibration factors (with the associated uncertainties) for the commercially available dosimeters used.

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# How the dosemeter's placement on protective apron may affect the $H_p$(10) measurements in interventional cardiological and radiological procedures

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The radiation protection and monitoring of medical staff in interventional procedure is a challenging task [1]. It is well known that the exposure of operators can be reduced by wearing protective equipment (aprons and collars) and by properly positioned shielding [2]. Notwithstanding that, due to the increasing number of yearly performed procedures, the annual personnel doses of these professionals are among the highest registered for workers in the medical field [3]. For instance, increasing the knowledge of the exposure condition and optimizing radiation protection is fundamental to fulfil the ALARA (As Low As Reasonably Achievable) concept for these workers.

In that context, the EURADOS (European Radiation Dosimetry Group) Working Group 12, devoted to dosimetry in medical imaging procedures, performed a series of investigations on the doses received by the medical staff during interventional procedures [4-5]. These studies showed clearly how the complexity of stray field reaching the operators can influence the doses to the operator.

For that reason, it was decided to conduct further analyses to determine the possible effects of that stray field on the registered doses using an apron-mounted dosemeter, which is frequently employed for operator dose assessment. The study has been performed through Monte Carlo simulations validated through measurements. It does not claim to identify the "best" position for the dosemter, but it is aimed at evaluating the variability of the responses for a whole body dosemeter placed at the operator's chest level, due to its position and to the position of the operator within the stray field.

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# VIRTUAL ESTIMATION OF EFFECTIVE DOSE IN NEUTRON FIELDS: APPLICATION OF THE 'PODIUM' APPROACH TO A SIMULATED WORKPLACE

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The CONCERT-funded project PODIUM [1] aimed to provide online personal monitoring of occupationally exposed workers, including real-time assessments of effective doses, by tracking the motion of individuals in workplace fields. One such set of workplaces are the mixed neutron-gamma fields that can exist within the nuclear industry or around accelerator halls, for instance.

To achieve this, a 'spectral' approach has been developed in which Monte Carlo modelling is used to pre-calculate a map of the radiation field as a discrete function of location and individual orientation (and time, if necessary), and where the fluence-energy distribution at each point is decomposed three-dimensionally into binned angle components. These components are then convolved with fluence to effective dose conversion coefficients within the Monte Carlo model, before being weighted and summed to build an effective dose rate map. By tracking the movement of individuals around this map using motion-capture camera technology, time integrated effective doses may be determined.

The approach adopted in this work is described, detailing the methods developed for generating and applying the effective dose rate map. A proof-of-concept of the approach is then demonstrated using a simple simulated workplace neutron field within a calibration laboratory, with corroborative comparisons made using measurements performed under controlled conditions. Those measurements included the use of a variety of passive and active instruments; good agreement was found overall, with any differences that were seen generally explainable and useful to provide insight into the accuracy of both the computational and physical dosimetry methods. Testing of the proposed real-time personnel dosimetry system was also demonstrated, via simulated tracking of an individual within the facility. A total effective dose of 1.25 μSv was recorded for a ~90 s exposure, accounting for dose rates as low as 0.5 nSv $h^{-1}$, which is much lower than anything that could be accurately measured by physical neutron dosemeters in such a field.

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# e-Butterfly: a cloud computing Industry 4.0 neutron spectrum unfolding code

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In this work, we present the e-Butterfly, a cloud computing Industry 4.0 neutron spectrum unfolding code that uses a recursive iterative procedure based on the SPUNIT algorithm [1]. The code was designed under the Python programming language, using Streamlit framework [2], and it is executed on the cloud, as Industry 4.0 cloud computing technology through Internet, by using mobile devices with Internet connectivity and a Web navigator. The code also employs python modules such as pandas, NumPy, Math, PlotlyExpress and xlsxwriter.
The user can freely choose the response matrix, number of detectors, and the initial spectrum, as well as, the conversion coefficients for a dosimetric quantity of his preference. There is no limit for energy bins or detectors. Alternatively, the user can choose an algorithm [3] for the selection of an Automated Initial Spectrum, obtained from the compendium on neutron spectra edited by the International Atomic Energy Agency, IAEA [4]. In this case, is necessary to use the same 60 energy bins from IAEA compendium. The code was validated using data from the IAEA compendium: Bonner spheres and response matrices from Physikalisch-Technische Bundesanstalt, PTB, ($^3$He) and Forschungszentrum für Umwelt und Gesundheit GmbH, GSF, ($^6$LiI(Eu)).
Results obtained show that the e-Butterfly unfolding code solves the neutron spectrum unfolding problem with high accuracy and it is easy, friendly, and intuitive. The code also shows plots and tables with descriptive statistics, helping the user to make a preliminary analysis of the unfolding process, before downloading the results. Furthermore, input and output data can easily be organized, according to user preference, in ".csv", ".txt" or ".xlsx" format. An important issue is that e-Butterfly also eliminates the drawback of requiring very expert users for their operation, once there is not the necessity to provide an initial guess spectrum for the deconvolution process.

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# Out-of-field doses in paediatric photon and proton radiotherapy - Summary of the EURADOS WG9 activity

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Since 2010 EURADOS Working Group 9 (Radiation Dosimetry in Radiotherapy) involved in investigation of secondary and scattered radiation doses in X-ray and proton therapy especially in case of paediatric patients.The main goal was as to analyse and compare out-of-field organ doses inside 5 and 10 year old paediatric anthropomorphic phanoms for a treatment of 5 cm diameter brain tumour. Proton irradiations were carried out at Bronowice Cyclotron Centre in IFJ PAN Krakow Poland using pencil beam scanning technique (PBS) at a gantry with dedicated scanning nozzle (IBA Proton Therapy System- Proteus 235). Gamma ray doses were measured with thermoluminescent dosimeters (MTS-7, MTS-6, MCP-n) and radiophotoluminescent detectors (GD-352M and GD-302M), while secondary neutrons were measured with two types of track etched detectors(PADC TypeI, PADC Type II) and with two types of bubble detectors (BDT, and BD-PND. Out-of-field doses measured for intensity modulated proton therapy (IMPT) were compared with previous measurementsperformed within WG9 collaboration for 3 different photon radiotherapy techniques: 1. Intensity Modulated Radiation Therapy (IMRT), 2. Three-Dimensional Conformal Radiation Therapy (3D CDRT) performed on a Varian Clinac 2300 linear accelerator (LINAC) in Centre of Oncology, Krakow, Poland, and 3. Gamma Knife Surgery performed on the Leksell Gamma Knife (GK) (Model 4 C, Elekta Instruments, Stockholm, Sweden) at the University Hospital Centre Zagreb, Croatia [1-3]. Phantoms and detectors used in all experiments as well as target location were the same for both photon and proton modalities.

Total secondary organ doses in proton therapy were found to be significantly lower (2-3 orders of magnitute difference) in comparison with the different photon radiotherapy techniques for the same delivered tumour dose. Moreover, for IMPT neutron doses are lower than non-neutron doses close to the target, but become larger than non-neutron doses further away from the target.

Results of WG9 studies provided out-of-field dose levels enquired for an extensive set of radiotherapy techniques, including proton therapy, and involving a complete description of organ doses of radiotherapy children patients. Such studies are needed both, for validating mathematical models of out-of-field doses and for epidemiological studies to allow risk estimation of second cancers and other late effects.

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# Passive dosimetry of secondary radiation produced by HYPERSCAN pencil scanning proton beam

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Mevion S250i Hyperscan gantry-mounted, compact proton therapy system is a newly developed radiotherapy solution. The first single room superconducting synchrocyclotron Mevion facility in Europe has been put into operation in Maastro clinic in Maastricht, Netherlands [1]. In this system 227 MeV pulsed proton beam (10 µs width and 750 Hz frequency) is delivered using spot scanning technique. Multileaf collimation system for dynamic layer-by-layer collimation and range modulation system for energy switching are mounted on the nozzle in the proximity of a patient.

EURADOS WG9 'Radiation Dosimetry in Radiotherapy' in an extensive experimental campaign characterized secondary radiation field for Mevion S250i Hyperscan synchrocyclotron. In phantom measurements with thermoluminescent, radiophotoluminescent, track-etched and bubble detectors were accompanied by measurements of stray radiation with active detectors inside the therapy room. The application of passive detectors overcome the problem of detector saturation by neutron and photon high dose-rates pulses induced by pulsed proton beam. 3D mapping of out-of-field doses was performed in a water phantom using sets of frames positioned in front, inside and behind the target volume of 10 x 10 x 10 $cm^3$. Each frame contained five tubes located above, inside and below the target, where detectors were inserted in particular positions. Subsequently, organ doses were determined during treatment of brain tumour with 5 cm diameter. We used two paediatric anthropomorphic phantoms of 5- and 10-years old children (type 705D and 706D, respectively) [2]. Out-of-field doses inside the water phantom were higher in comparison to corresponding irradiations for a dedicated proton scanning nozzle [3]. On the other hand, out-of-field doses measured for compact Mevion facility are significantly lower with respect to standard photon therapy techniques. [4]

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# Joint EURADOS WG9-WG11 rem-counter intercomparison in a Mevion S250i Hyperscan pulsed stray neutron field: lessons learnt and comparisons with other proton and photon therapy facilities

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Proton therapy is gaining popularity because of the improved dose delivery over conventional radiotherapy. The secondary dose to healthy tissues is dominated by stray neutrons [1]. In the past few years, the European Radiation Dosimetry Group (EURADOS) has organised several measurement campaigns with the goals of understanding the limits of neutron dosimetry in proton therapy, promoting good practices and providing guidelines. Rem-counters are robust instruments for the on-line assessment of neutron ambient dose equivalent (H*(10)). In general, however, *a priori* knowledge of the facility and of the neutron field is required for the choice of any survey meter. A forthcoming paper [2] reports on an extensive rem-counter intercomparison at the Maastro Proton Therapy centre (Maastricht, NL). The centre is a single-room facility using the compact Mevion S250i Hyperscan synchrocyclotron. The accelerator provides a scanned 227 MeV proton beam, delivered in pulses with 10 µs width and 750 Hz frequency, which is afterwards degraded in energy by a passive range modulator [3]. Several rem-counters were tested (LUPIN, LINUS, WENDI-II, LB6411, NM2B-458, NM2B-495Pb) by simulating a patient treatment using a water tank phantom. Only the LUPIN, which is specifically designed for pulsed fields, correctly assessed H*(10). All other rem-counters underestimated H*(10) by factors from 2 to more than 10, depending on the detector model and on the neutron dose per pulse. In pulsed fields, the neutron dose per pulse is a fundamental parameter, while the average dose rate is rather a secondary quantity [4]. For this reason, rem-counters that proved to be adequate at non-pulsed facilities [5] here drastically underestimated H*(10). In this work, we report the main results and the lessons learnt from the Maastro campaign. We also compare Maastro H*(10) (150-200 µSv/Gy at 2-3 m from the phantom) with data from other proton therapy facilities, with a focus on active energy selection systems. Finally, H*(10) is compared with results from conventional radiotherapy facilities. We found that H*(10) at Maastro is about ten times higher than other systems employing active energy selection. On the opposite, in some cases of purely passive systems, as well as of conventional 15 MV radiotherapy, the Maastro H*(10) was about ten times lower.

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# Relative efficiency of radiophotoluminescent glass detectors in low energy proton beams

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In charged particle dosimetry luminescence efficiency strongly depends on linear energy transfer (LET). Therefore, for correct interpretation of the measurements in hadron radiotherapy or space dosimetry it is important to characterize the detectors efficiency as a function of LET. Unlike for other luminescent detectors, available literature data for the efficiency of radiophotoluminescent (RPL) glass detectors are scarce and especially for low energy range no data was found. The aim of this study is to evaluate relative efficiency of RPL glass detectors in low energy proton beams.
RPL glasses of environmental dosimeters (silver activated phosphate glass FD-7/SC-1; AGC Techno Glass Co.) with dimensions 16 x 16 x 1.5 $mm^2$ were used in this study. Measurement of the RPL signals were carried out with the FDG-202 reader. Detectors were exposed to proton beams in a high vacuum chamber at the microprobe experimental line of the RBI accelerator facility. A focused proton beams ($1 \times 1$ $mm^2$) were delivered using two electrostatic tandem accelerators: 1 MV and 6 MV accelerator for up to 2 MeV and 3-5 MeV proton beams, respectively. In all cases protons were stopped in the detectors. The range and LET of protons in water and RPL glass were calculated using SRIM code and used for absorbed dose calculations. Detectors were exposed to protons at the fluence range of $3.3\times10^6$ to $1.3\times10^8$ p/$mm^2$ while doses absorbed in the irradiated volume, expressed as dose to water, ranged from 7.6 to 797.8 Gy. Firstly, fluence/dose response was investigated [1]. Secondly, RPL signals from the linear dose response range were used to evaluate efficiency relative to reference gamma radiation.
Preliminary results show that in the linear fluence/dose response range the mean RPL signal normalized to the number of incident protons is decreasing exponentially with decrease of the beam energy. The relative efficiency (expressed for dose in water, relative to $^{60}$Co) is also decreasing with energy/LET from 0.6 for 5 MeV (8.1 keV/$\mu$m) and 0.2 for 3 MeV (12.1 keV/$\mu$m) to approximately 0.1 for 2 MeV and below.

This work is supported by the Croatian Science Fundation (Project IP-02-2020-3593)

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# A spectrometry study of a pulsed stray neutron field in Mevion S250i Hyperscan Proton Therapy Facility

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Proton beams enable a highly conformal and biologically effective therapy dose delivery within the tumour volume. However, some concerns remain due to long-term side effects following radiation therapy which are related to the dose deposited in the healthy tissue surrounding the tumour. The major contribution to the dose in out-of-field regions is delivered by secondary neutrons, which are produced by nuclear interactions of the primary protons with the beam line components, the patient's tissue and the components in the treatment room. Consequently, the measurement of stray neutrons in the treatment room around phantoms irradiated with protons is a very important first step for estimating health risks of developing side effects, for example secondary cancer.

Recently, Working Group 9 'Radiation dosimetry in radiotherapy' of the European Radiation Dosimetry Group (EURADOS) carried out a large measurement campaign at the Maastro Proton Therapy centre in Maastricht, Netherlands. This single-room facility is equipped with the compact Mevion S250i Hyperscan synchrocyclotron providing a scanned proton beam with energy up to 227 MeV, delivered in pulses with 10 µs width and 750 Hz frequency, which is afterwards degraded in energy by a passive range modulator mounted on the nozzle.

In this study we report on neutron spectrometry and dosimetry using extended range Bonner Sphere Spectrometry (ERBSS) systems. The secondary neutron spectra were measured at different positions around a 10-year old anthropomorphic phantom in the treatment room. Measurements were supplemented by Monte Carlo simulations of the radiation field in the treatment room. The impact of distance to isocenter and angle with respect to beam direction on the shape of neutron spectra, fluence rates and ambient dose equivalent values, H*(10), was studied. Interestingly, the highest neutron H*(10) value of about 30 µSv/Gy was assessed at rear positions with respect to beam direction close to the synchrocyclotron at a distance of about 2 m from the isocenter, while about 20 µSv/Gy was measured at positions perpendicular to the beam direction at a distance of about 2 m from the isocenter.

Neutron spectra measured at Maastro were also compared to neutron spectra at Cyclotron Centre Bronowice (CCB) in Kraków, Poland, that had been measured at similar positions around the same 10-year old anthropomorphic phantom, but for a system with dedicated scanning nozzle [1]. We found that H*(10) values at Maastro were about ten times higher than those measured at CCB.

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# Passive detectors' response to mixed fields from the CERF facility for their use in proton therapy out-of-field dosimetry

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The Working Group 9 (WG9: Radiation dosimetry in Radiotherapy) of the European Radiation Dosimetry Group (EURADOS) has organised several measurement campaigns and detector intercomparisons in the last years with the aim of characterising out-of-field doses in proton therapy [1,2]. Out-of-field dosimetry in this condition is complex due to the presence of different types of radiation and of high energy neutrons, so that it is not straightforward to assess total dose from measurements.

The CERN-EU high-energy Reference Field (CERF) [3] is a calibration facility where a well characterized mixed radiation field of neutrons, photons and charged particles is produced. The neutron energy spectrum at CERF is similar to that present in proton therapy facilities, so that it is possible to characterize the response of passive detectors in a mixed reference field with characteristics comparable to those of the places where they are used.

An irradiation campaign was organized during the summer 2021 CERF run using several types of passive detectors: Thermoluminescence detectors (TLDs), with Li natural composition (MTS-N, MCP-N), Li-7 enriched TLDs (MTS-7, MCP-7) and Li-6 enriched TLDs (MTS-6, MCP-6); Radio-photoluminescence detectors (RPLs), types GD-352M and GD-302M; and PADC (Poly-Allyl-Diglicol-Carbonate) etched-track detectors. One subset of TLDs was enclosed in Teflon holders, while another was positioned in a PMMA holder to investigate the potential effect of the holder on the detector response. RPLs were irradiated without and with a PMMA cover. Finally, PADC detectors were placed on PMMA holders and covered with different sets of converters, with diverse combinations of Nylon, Makrofol, Polyethylene and Lead. Irradiations took place at the Concrete Top 2 (CT2) and Concrete Side 1 (CS1) reference positions of the CERF facility [3].

The response study from this work, resulting in improved calibration coefficients that take into account the mixed photon and neutron field with a high energy neutron component, like that present in proton therapy, will allow to make better assessments of out-of-field doses.

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# Gafchromic in-vivo dosimetry for dynamic conformal arc therapy.

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The motivation for this work was to develop a method for performing in vivo dosimetry for patients with unresectable sarcomas treated with dynamic radiotherapy techniques.

The primary goal of the study was to evaluate the use of Gafchromic films for monitoring the doses received by patients with unresectable sarcomas who underwent preoperative hypofractionated radiotherapy 5x5Gy. The secondary goal was to evaluate the quality of radiotherapy in this group of patients. The consistency of the dose measured during the therapeutic session and the dose calculated in the treatment planning system, as well as the repeatability of a dose delivered during the therapeutic session, were evaluated.

The validation of Gafchromic films measurement method was performed and its uncertainty was estimated on the basis of phantom measurements. The uncertainty of determining the reference dose, obtained from the treatment planning system in measurement points, was estimated. In vivo measurements were performed for 21 patients. The results obtained in 96 therapeutic sessions (760 EBT3 / EBT-XD film detectors) were analyzed.

The difference between the measured dose and the reference dose was determined. A procedure algorithm was defined for results exceeding the action level (7%). Statistics describing the repeatability of the treatment for 21 patients at each measurement point (4 measurement points for each patient) were calculated.

The results of the study showed that Gafchromic films may be used for in vivo dosimetry for dynamic techniques for patients with unresectable sarcomas who underwent preoperative hypofractionated radiotherapy. The low uncertainty (not exceeding 1,1% - one standard deviation) of Gafchromic films measurement provides the detection of errors that are considered dangerous for the patient undergoing radiotherapy. In 93% of the cases, the difference between the measured dose and the reference dose did not exceed 7%. The results of the study reviled compliance of the delivered doses with the doses calculated in the treatment planning system and a good repeatability of treatment.

The proposed method of in-vivo dosimetry carried-out with Gafchormic films is a promising method of verification of dose delivered to patients treated with the dynamic conformal arc therapy irradiated with a bolus. The routine use of in vivo dosimetry with Gafchromic films requires the development of a simple method of determining additional dose from patient imaging and automation of dose readout from the treatment planning system.

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# Comparing the effects of various (p,xn) production on the characteristics of secondary neutrons and potential induced radioactivity in concrete walls of a biomedical cyclotron

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There are currently 13 medical isotope production cyclotrons in Taiwan. The proton energies of these cyclotrons are between 9.6–30 MeV. The following (p,xn) nuclear reactions with various targets are involved in their routine operations for medical isotope production: $^{18}O(p,n)^{18}F$, $^{68}Zn(p,n)^{68}Ga$, $^{68}Zn(p,2n)^{67}Ga$, $^{69}Ga(p,2n)^{68}Ge$, $^{100}Mo(p,pn)^{99}Mo$, $^{100}Mo(p,2n)^{99m}Tc$, $^{89}Y(p,n)^{89}Zr$, $^{112}Cd(p,2n)^{111}In$, $^{203}Tl(p,3n)^{201}Pb$, $^{124}Xe(p,2n)^{123}Cs$, and $^{15}N(p,n)^{15}O$. Secondary neutrons accompanying these production routes could induce material activation in the cyclotron body and its surroundings to varying degrees. Considering a simplified model of a biomedical cyclotron facility with the capability of implementing these production routes, the authors evaluated the characteristics of secondary neutrons and potential induced radioactivity in surrounding concrete walls by using the FLUKA Monte Carlo code. FLUKA is not only a particle transport Monte Carlo code, but also an integrated system for the buildup and decay of produced radioisotopes. In this study, simulating the proton bombardment of various targets, angular and energy distributions of neutrons accompanying these production routes were scored and compared. In addition, the subsequent transport and interaction of these neutrons in the cyclotron vault was followed, and then the accumulation and decay of neutron-induced radionuclides in various materials can be estimated. Residual long-lived radioactivity in forward and lateral concrete walls of the cyclotron is of most concern because their quantities and trends are closely related to decommissioning planning for a cyclotron-based radionuclide production facility. Among these production routes, the activation results induced by neutrons from the $^{18}O(p,n)^{18}F$ reaction with 18 MeV protons were taken as the reference, from which the results with other targets and proton energies were compared. Accordingly, the effects of various (p,xn) production routes on the induced radioactivity in concrete walls of a biomedical cyclotron can be quantified. Detailed results will be presented and discussed at the conference.

# The GEMTEQ: A GEM Based Detector with Highly Pixelated Readout for Microdosimetry and 3D Particle Track Reconstruction

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The GEMTEQ is a small gaseous detector based on the GEMPix [1], which is constructed through coupling two CERN technologies, Gas Electron Multipliers (GEMs) [2] and Timepix chips [3]. Microdosimetric measurements are possible due to the high granularity of the four Timepix chips: this readout features 512 x 512 pixels with a pitch of 55 µm. Whereas the GEMPix is operated with the standard gas mixtures used for GEMs, typically $Ar:CO_2$ or $Ar:CO_2:CF_4$, the GEMTEQ works with tissue-equivalent (TE) gas at atmospheric pressure, and critical components such as the radiation window are built from TE materials.

Measurements in the mixed radiation field of an AmBe source were performed to prove the microdosimetric capabilities of the GEMTEQ. A standard microdosimetric dose spectrum was obtained. Furthermore, a neutron/photon discrimination method is demonstrated based on the pixelated nature of the readout system.

Conventional Tissue Equivalent Proportional Counters are usually operated at low pressures such that the mean chord length in their active gas volume is equivalent to typically one or two micrometres in tissue. Recently, a vacuum chamber has been designed, assembled and tested at CERN to house the GEMTEQ for sealed and low-pressure operation. The rationale being: (1) to enable measurements in places, where it is impossible to flush a detector continuously with tissue equivalent gas and (2) to operate the GEMTEQ at pressure values lower than atmospheric pressure, which increases the spatial resolution. Tests of the chamber showed that the GEMTEQ can be sealed and operated in the chamber for at least one week without signal deterioration. Currently, the detector is working well at pressures as low as 200 mbar.

The operation of the GEMTEQ inside the newly constructed vacuum chamber has been tested in clinical carbon ion and proton beams at the Italian National Centre for Oncological Hadrontherapy (CNAO). In this measurement campaign, the GEMTEQ was operated as a time projection chamber and, knowing the electron drift velocity, 3D particle tracks were reconstructed. The tracks have been analysed and LET values were calculated. This demonstrates the potential of the GEMTEQ to reach a performance beyond standard experimental microdosimetry.

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# Light output of 3D printed plastic scintillators after surface finishing and wrapping


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Plastic scintillators are commonly used in radiation detectors in medical physics applications [1]. Digital light processing (DLP) 3D printing is a novel method for manufacturing of plastic scintillators allowing unlimited shaping in a relatively short time [2].

The light output of 3D printed plastic scintillators with the dimension of 10 mm × 10 mm × 10 mm emitting blue light [3] was investigated, applying nine configurations of surface finishing and wrapping, and were compared to the original samples manufactured by the DLP 3D printing machine. The sample surfaces were finished with sandpaper and/or polishing paste, and/or wrapped with optical reflectors, i.e., the white polytetrafluoroethylene (PTFE) tape and the Enhanced Specular Reflector (ESR) foil. Samples, exposed to the Cs-137 source, were connected to vacuum photomultiplier tube (PMT) that collected the light produced by gamma radiation in the scintillator. Depending on the surface manufacturing method the amount of light reflected on the sample walls and entering PMT varied, leading to different light output signals.

The results showed that the light signal detected with the PMT increased by a factor of about 2.6 for wrapped and sanded scintillators and 2.1 for wrapped and polished scintillators with respect to the original samples. The difference between the light signal collected from scintillators wrapped with PTFE tape and ESR foil is about 1%. In scintillators with no wrapping, about half of the emitted light escaped through side walls and top of the sample cube. There is no substantial difference between the light signal collected from the original sample and sample with finished walls, allowing assembly of any detector from a 3D printed scintillator without additional manufacturing. We demonstrated that surface finishing and wrapping improves light output of 3D printed scintillators enabling its application as dosimetry material.

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# Characterization and comparison of cell casted and 3D printed plastic scintillators for dosimetry applications

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Plastic scintillators emit light when irradiated by ionizing radiation and are characterized by numerous superior dosimetric properties, such as: fast timing [1], high optical transmission [2], possibility of manufacturing of large active volumes without substantial increase of cost. Currently, the most used methods of plastic scintillator manufacturing are cell casting and bulk polymerization, extrusion, injection molding [3], while digital light processing (DLP) 3D printing technique has been recently introduced [4].

We compared light output of commercial, cell casted blue-emitting RP-408, EJ-200, EJ-208 and green-emitting EJ-260, EJ-262 scintillators and two types of blue-emitting DLP 3D printed scintillators, with wavelength of maximum emission ranging from 425 nm to 490 nm. Cell casted scintillators were manufactured from polyvinyltoluene and DLP 3D printed scintillators were made from bisphenol A resin type as a polymeric base. All samples had the same cubic shape with dimensions 10 mm x 10 mm x 10 mm and were exposed to the Cs-137 source. These samples were connected to a vacuum photomultiplier tube (PMT) that collected the light produced by gamma radiation in the scintillator.

Relative light output of two types of 3D printed scintillators in comparison to blue-emitting reference cell casted scintillators was about 50%. Measured light output of green-emitting cell casted scintillators was about 30% lower than for blue-emitting cell casted scintillators. Performance of the 3D printed and green-emitting scintillators is sufficient to use these materials in a plastic scintillation dosimeter operating in high fluence gamma radiation fields. Also, we found that plastic scintillators emitting green light are more resistant to radiation damage and emit less Cherenkov stem signal during irradiation, with respect to the scintillators with shorter wavelength.

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# Shielding analysis of the proton therapy facility at China Medical University Hospital: comparison of a simplified point-source line-of-sight model with Monte Carlo simulations

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As per Particle Therapy Co-Operative Group (PTCOG) statistics, Taiwan will have 10 particle therapy facilities, with two in operation and eight to go, the highest density in the world. During the operation of particle therapy facilities, secondary radiation is significantly produced at locations where beam losses occur. The shielding design of these facilities usually involves meters thick concrete walls and sometimes supplemented by additional local shielding for hot spots. The induced radiation field in these facilities could be complex depending on the beam energies, target materials, beamline designs and facility layout. Monte Carlo simulations are generally considered the most accurate method for such a complex accelerator shielding analysis. However, full-scale Monte Carlo simulations are inevitably time-consuming, both in model building and computation. Therefore, simplified shielding models based on point-source line-of-sight approximation are occasionally preferable in practice because of ease of use and intuitive understanding. Taking the shielding analysis of the proton therapy facility at China Medical University Hospital as an example, this study performed a systematic and in-depth comparison between the two approaches.

The proton therapy facility at China Medical University Hospital has a 250 MeV proton cyclotron and two gantry rooms. For each beam loss scenario, two full-scale Monte Carlo shielding simulations were performed using two independent Monte Carlo codes, FLUKA and PHITS, respectively. The agreement between the two codes was good, which in turn confirmed the validity of the two Monte Carlo simulations. The Monte Carlo results were therefore taken as the reference from which the accuracy and reliability of the simplified model prediction can be evaluated. The simplified model adopted in this study was essentially a point-source line-of-sight model with shielding parameters (source terms and attenuation lengths) suggested by *Sheu et al.* (2013) for estimating dose rates behind thick shields at proton therapy accelerators. For dose rates at locations near the maze exit, an appropriate coupling between the unshielded source term and empirical formulae for radiation streaming was used instead. Focussing on this real-world shielding problem, the authors estimated dose rates at many locations of interest around the whole facility by using the proposed model and Monte Carlo simulations. The aim of the study is to examine the general applicability and limitations of the simplified model in complex shielding problems of proton therapy accelerators. Detailed results will be presented and discussed at the conference.

# 60 MeV proton FLASH beam line at IFJ PAN in Kraków for research in radiobiology and dosimetry

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The introduction of the FLASH technique to proton radiotherapy requires the development of appropriate dosimetric methods and conducting in-depth radiobiological studies. The aim of the work was to test beam structure and dosimetric properties of the newly configured high-intensity proton beam facility at the AIC -144 cyclotron at IFJ PAN, Kraków.

The radiation field size was verified using Gafchromic EBT3 films and independently with 2-D MCP-N (LiF: Mg,Cu, P) thermoluminescent foils developed at IFJ PAN. The beam dosimetry was performed using a set of alanine pellets and LiF:Mg,Cu,P ( MCP-N) TLD detectors placed in a specially prepared PMMA phantom. The measurements of the dose rate in the beam macro pulse were carried out with the newly developed RL/OSL PORTOS dosimetric system with the sampling time between 1μs equipped with a slice (1 mm in thickness) of a $LiMgPO_4:Tm$ crystal.

Measurements with all dosimetric methods demonstrated consistent results. FLASH dedicated irradiation unit allows to perform exposure with 50 MeV protons, 20 mm beam diameter and uniformity better than 10%,. Proton beam accelerated in the AIC-144 cyclotron shows a double beam structure, i.e. the micro-pulses related to the RF frequency of the cyclotron and the macrostructure where the micro-pulses are grouped into packets with a length of 0.5 ms and frequency of 50 Hz. The average dose rate was exceeding 40Gy/s in a field with a diameter of 20 mm but the dose rate in the macro-pulses was estimated at 1600 Gy/s.

The facility at the AIC-144 cyclotron is available for hosting of FLASH irradiation of biological cells, tissues and small animals. We also offer reliable, active and passive methods for FLASH dosimetry.
The study was partially supported by the Horizon 2020 project INSPIRE Grant No. 730983.

# Alzheimer Disease and NECTAR project: characterization of a neutron facility for brain cell irradiations

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Alzheimer Disease (AD) is the most common cause of dementia [1]. Most disease-modifying treatments are beta amyloid peptide (A-beta, Aβ)-centric based on a strong faith in the Aβ Cascade Hypothesis and they are based on a drug-mediated targeting of amyloidogenic and non-amyploidogenic pathways [2].
NECTAR (NEutron Capture-enhanced Treatment of neurotoxic Amyloid aggRegates), a project funded by the EU H2020 programme (FET Open call), proposes a novel strategy to address AD treatment, suggesting and investigating a therapy based mainly on neutron capture nuclear reactions on $^{10}$B and $^{157}$Gd and structural damage of AD Aβ aggregates induced by generated secondaries. More specifically, NECTAR aims to develop, test and prove the feasibility, safety and effectiveness of a Capture-Enhanced Neutron Irradiation of Aβ aggregates exploiting the synergy between (*i*) an external beam of low energy neutrons which irradiate the whole brain and (*ii*) specifically engineered radiation enhancers capable of increasing the administered dose in the Aβ aggregate site only.
A first key step of the project is the set-up of a well-characterized neutron facility where performing irradiations of cells doped with novel radiation enhancers. The collimated thermal neutron beam designed and developed at the LENA TRIGA Mark II reactor of Pavia was selected as the test facility of the project. An extended experimental activity was then carried out to characterize in detail the beam through (*i*) neutron spectrometry and (*ii*) microdosimetry. The former was performed by mean of activation foils and with the single-moderator active neutron spectrometer DIAMON [3], the latter with the avalanche Tissue Equivalent Proportional Counter developed by Politecnico di Milano [4].
The present work discusses the results of the whole characterization campaign.

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# Applications of Nanodosimetry in Particle Therapy Planning

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In this talk, I will give an overview of application of nanodosimetry in particle therapy treatment planning [1]. My talk will summarize the underlying concepts of nanodosimetry and describe the development and current status of nanodosimetric detector technology. I will also give an overview of Monte Carlo track structure simulations that provide nanodosimetric parameters for proton and ion therapy treatment planning. Classical and modern radiobiological assays that can be used to demonstrate the relationship between the frequency and complexity of DNA lesion clusters and nanodosimetric parameters will be reviewed. Lastly, I will review existing approaches of treatment planning based on RBE models or dose-averaged linear energy transfer and contrast them with the RBE-independent approach based on nanodosimetric parameters.

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# Single crystal diamond-based microdosimeters for ion-beam therapy applications

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**Purpose**: Due to its outstanding properties, diamond has been considered a suitable material for the fabrication of high-spatial resolution and radiation hard detectors for a wide range of beam qualities used in radiation therapy, including ion beams. Microdosimeters based on synthetic single crystal diamond with a micrometric sensitive volume were recently developed at University of Rome "Tor Vergata" laboratories, showing very promising capabilities for the characterization therapeutic ion beams [1-3]. In the present work, we report a comprehensive investigation of the microdosimetric properties of such diamond detectors under different ion beams and microbeams.

**Materials and Methods**: Several diamond-based microdosimeters were realized in highly doped / intrinsic diamond / Schottky contact configuration by means of both standard photolithography and selective chemical vapor deposition techniques to precisely define their sensitive volume. Ion beam induced charge (IBIC) technique was at first performed using different particles microbeam to characterize the response of the microdosimeters in terms of their charge collection properties. Microdosimetric measurements were then carried out at different depths both at the scanned carbon-ion beam of the CNAO and at the 62 MeV proton beam of CATANA. Experimental data were also compared with those obtained with a miniaturized Tissue Equivalent Proportional Counter (mini-TEPC) used as reference.

**Results**: Diamond-based microdosimeters exhibited a well-defined micrometric active volume and good spectroscopic properties. A good agreement between relative biological effectiveness (RBE) values calculated with diamond and the mini-TEPC was observed. The experimental results showed a great potential of such devices for the microdosimetric characterization of therapeutic proton and carbon-ion beams, indicating their suitability for quality assurance of radiation quality in ion-beam radiation therapy.

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# Microdosimetric in- and out-of-field radiation field characterization of proton, helium and oxygen beams.


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Microdosimetry takes into account stochastic effects of energy deposition, and for this reason can offer a superior characterization of the radiation field quality compared to standard dosimetry. This approach is widely used in particle therapy for predicting the cell survival and assess the RBE.

In this work, we used microdosimetry to investigate the in- and out-of-field radiation field in irradiations with proton, 4-Helium and 16-Oxygen beams. Protons are the most common ion type used in the clinics, while the first patient has been treated at HIT (Germany) with 4-Helium. 16-Oxygen represents a very promising candidates for radioresistant tumors, for which an ion heavier than 12-Carbon is required.

The microdosimetric spectra were acquired with a Tissue Equivalent Proportional Counter (model LET ½ Far West Technology) positions at different lateral positions and depths inside a water phantom. The experiments with protons were performed both with a 148 pencil beam [1] and a Spread-Out-Bragg-Peak (SOBP) [2] at the Trento Proton therapy Center (Italy). The measurement campaigns with 160 MeV/u 4-Helium and 360 MeV/u 16-Oxygen pencil beams were carried out at the NASA Space Radiation Laboratory (NSRL) of Brookhaven National Laboratory (US).

While the in-field data can be used to validate radiobiological models for tumor control probability, the out-of-field data can help assessing the irradiation of normal tissue, from which potential toxicity effects are induced.

In all experiments, we collected yd(y) spectra, and calculated the distribution parameters ($y_F$, $y_D$ and $y^*$). In addition, at several depths in- and out-of-field, we estimated the Relative Biological Effectiveness (RBE) values exploiting the Microdosimetric Kinetic Model (MKM), and the quality factor Q according to the recommendation of the ICRP-ICRU.

We further simulated all experimental data with the microdosimetric extension of TOPAS Monte Carlo code [3]. The comparison includes the yd(y) distributions, as well as the RBE and Q profiles, which are in good agreement for protons, but indicate relatively large discrepancies for the heavier ions. At last, we calculated the contribution of each particle species to the overall microdosimetric spectra and to the overall quality factors for the most interesting out-of-field positions.

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# A wall-less TEPC for nanodosimetric measurements

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An ever deeper understanding of the interaction between ionizing radiation and soft tissue is of primary importance for radiation therapy and, in particular, for hadron therapy. These treatments exploit charged hadron beams to cause severe damage in the DNA of malignant cells and inactivate them. It is recognized that the observable radiobiological effects of ionizing radiation are strongly correlated to the clustering of damages in micrometer- and nanometer-sized subcellular structures, hence to the particle track structure. The characteristic properties of track structure are directly measurable nowadays with bulky experimental apparata (i.e. nanodosimeters), which cannot be easily operated in a clinical environment. With this in mind, a wall-less avalanche-confinement Tissue Equivalent Proportional Counter (TEPC) was properly designed for measuring microdosimetric distributions down to the nanometric region directly within the same chamber of the STARTRACK nanodosimeter, installed at the Legnaro National Laboratories (INFN). This set-up allows a direct comparison between the spectra acquired by the TEPC and the corresponding nanodosimetric probability distribution measured by the nanodosimeter [1].

A study about experimental gains of the filling gas of the detector has been conducted aiming at verifying the agreement with theoretical gains evaluated through different models. The TEPC was irradiated with an isotopic $^{244}$Cm alpha source at different impact parameters (i.e. distances between its center and the trajectory of the primary particle) showing that, once calibrated, it is capable of assessing a nanodosimetric cluster size distribution for some tens of nanometers in simulated size. Several Monte Carlo simulations have been performed, showing a good agreement with experimental results.

Finally, cluster size distributions produced by $^{244}$Cm alpha particles and 26.7 MeV $^{7}$Li ions were simultaneously measured with the STARTRACK nanodosimeter and with the avalanche confinement TEPC. The comparison of pairwise measurements shows an excellent agreement when the signals from the TEPC are processed to include the reduced detection efficiency and the dead time of the STARTRACK detector. This result suggests a minor influence of the gas-avalanche stochastics on the measured distributions and encourages the use of such proportional counters for the experimental characterization of therapeutic hadron beams at the nanometer level. The direct comparison between nanometric-domain microdosimetric distributions and track-nanodosimetric measurements supports the application of a nano-microdosimetric spectrum to derive track structure quantities at the nanometer level (for instance the mean cluster size M1 and the cumulative distributions F1, F2 and F3) which have been observed to correlate strongly with the radiation damage [2].

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# A novel microdosimetric model of clonogenic survival: benchmark against *in vitro* data and *in silico* fragmentation studies

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Accurate, robust and flexible models of the radiation-induced cell-death are of primary importance when planning external radiotherapy treatments with charged particles [1]. Therefore, many biophysical models of cell survival were developed over the years (see overview in [2]). Generally speaking, these models can be classified in two main groups: the ones based on track structure theory or on microdosimetry. The latter models have the advantage of dealing with experimentally measurable quantities and explicitly taking into account the stochastic energy deposition at the microscopic scale [3].

The most widely used microdosimetric model of clonogenic survival is the modified microdosimetric kinetic model (MKM) [4], which is also clinically implemented in Japan. Compared to the original MKM [5], the modified MKM [4] accounts for the well-known overkill effects by means of a phenomenological quantity (saturation-corrected dose-mean lineal energy, $y^*_D$) used in the calculation of the liner term (α) of the linear-quadratic model (LQM). In MK-based models, it is generally assumed that the quadratic term of the LQM (ß) is radiation-independent. This appears to be in contrast with recent experimental and theoretical results [6]. Furthermore, it was shown [2] that the modified MKM has relevant limitations in reproducing the *in vitro* RBE data for $^{20}$Ne and heavier ions.

Thus, corrections to the modified MKM were proposed by including dose-dependent microdosimetric distributions at both the nuclear and subnuclear scale in the mathematical formalism (stochastic- and double-stochastic MKMs from [7]). However, these models were found to be computationally demanding and work is ongoing to develop simplified approaches for their implementation into treatment planning systems [8].

This work introduces an alternative approach to quickly calculate the α and ß terms of the LQM by combining single-event microdosimetric distributions with multi-hit Poisson theory derived from early works with radiation detectors [9]. The results are presented as a function of the particle type, the LET, the survival fraction for different repair competent or deficient cell lines after single-dose exposures. A comparison with corresponding published *in vitro* clonogenic survival data and published track-structure DNA fragmentation studies is included, as well as a possible extension of the model formalism to account for altered cellular oxygen concentrations (hypoxia).

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# A new microdosimeter detector based on Silicon Carbide technology

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Silicon Carbide (SiC) has been recently proposed and patented as a material for the realization of radiation dosimetry and microdosimetry detectors [1,2]. SiC properties, including its high sensitivity and radiation damage resistance make it an amazing device susceptible to radiation dosimetric applications of direct ionizing radiations.
The SiC detector presented in this work has been designed by IMM-CNR and manufactured by ST-Microelectronics company (Production site of Catania, I). It has a 0.3 um thick p-layer with a doping concentration NA = 10^19 cm^3 and a 10-um thick n-layer with a doping concentration ND = 0:5–1 10^14 cm^3. The detector has an active thickness of 10 um and was mounted on a PCB board housed in an aluminium box. Depletion is obtained already at 10V while it was operated in an over-depletion condition, applying a 50V positive bias. The voltage applied is about 10 times higher than the depletion voltage to ensure a maximum drift velocity, necessary for a fast and complete charge collection.
The SiC detector was irradiated in two different facilities: at the CATANA (Centro di Adroterapia e Applicazioni Nucleare Avanzate) facility (IT) with 62 MeV clinical proton beams and with 35 MeV proton beam at Ústav Jaderné Fyziky Av Čr (CZ). The experimental measurements were performed at different depths to reconstruct the microdosimetric spectra along entire the Bragg peak. The depth was varied by inserting calibrated PMMA layers between the beam collimator and the microdosimeter. A total of 10^6 events were recorded at each measurement position. The same measurements were also performed by using a 10 μm thick SOI microdosimeter MicroPlus-Bridge designed by the Centre for Medical Radiation Physics (CMRP), University of Wollongong, Australia. The Bridge microdosimeter consists of an array of 3D right parallelepiped (RPP) shape SVs with area 30 μm x 30 μm each, constructed on a silicon-on-insulator (SOI) wafer with an active silicon thickness of 10 μm [3]. Both experimental set-up as well as the SiC detector was simulated with the Monte Carlo code Geant4.
The comparison between the two adopted detectors, as well as the good accordance with the simulation results, shown as the SiC is a promising material for micorodsimetric purposes. SiC exhibited a good energy resolution as well as a high radiation hardness. The experimental results showed a great potential of such devices for the microdosimetric characterization of therapeutic proton beams.

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# Microdosimetry of a 62 MeV clinical proton beam with five detectors

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**<u>Purpose</u>**: To introduce microdosimetry for the quality assurance in proton radiotherapy, different detectors have been developed. In this work, we characterized lineal energy distributions *yd(y)* acquired with five microdosimeters when exposed to the same 62 MeV modulated proton beam. The detectors used were the FWT LET-½, a miniaturized Tissue Equivalent Proportional Counter (mini-TEPC), a diamond and two silicon microdosimeters.

**<u>Materials and Methods</u>**: Measurements were performed in the 62 MeV Spread Out Bragg Peak of CATANA, at the same positions across the depth-dose profile. The FWT LET-½ has a spherical sensitive volume (*SV*) of 12.7 mm diameter and was filled with propane gas to simulate a site size of 2 µm [1]. The mini-TEPC has a cylindrical *SV* of 0.9 mm in diameter and height and was filled with propane to simulate a site size of 1 µm [2]. The novel synthetic single crystal diamond microdosimeter has an active planar-sectional area of 100 µm × 100 µm and a thickness of approximately 6.3 µm [3]. The monolithic silicon telescope has a 500 µm thick E-stage and a pixelated ΔE stage, with each pixel being a cylindrical diode of 1.9 µm thickness and 9 µm diameter [4]. The MicroPlus-Bridge microdosimeter has an array of 3D right parallelepiped shape *SV*s with area 30 µm × 30 µm each, and 10 µm thickness [5].

**<u>Results</u>**: Discrepancies in the microdosimetric spectra are analysed in terms of different material and geometry of the detectors. Despite the observed differences in the *yd(y)* distributions, the microdosimetric assessments of relative biological effectiveness (RBE) are in good agreement with each other and with radiobiological data.

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# BNCT@CNAO: a multidisciplinary research collaboration project

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CNAO Foundation will team with INFN, Polytechnic of Milan and University of Pavia (the BNCT@CNAO Project) to introduce, for the first time in Italy, a new particle accelerator for the production of neutron beams for Boron Neutron Capture Therapy. The accelerator is realized by the US company Tae Life Sciences and will be installed at CNAO, Pavia (Italy), in an area dedicated to clinical and medical research.

Pavia has already been a pioneer in this technique, thanks to the work carried out in the early 2000s by experts from the Policlinico San Matteo, the University of Pavia and the National Institute of Nuclear Physics (INFN). This wealth of experience will find new life in the project BNCT@CNAO. Two irradiation rooms will be built and served by a tandem proton accelerator: one devoted to research and the other reserved to patients treatment. Treatment rooms include fixed beam line, beam shaping assembly and a ceiling-mounted robotic couch for optimal patient positioning.

Due to its characteristics, BNCT requires multidisciplinary research, involving physicists and engineers for the design and implementation of the technology necessary for the production of neutron beams; chemists and biologists for the study and optimization of the bio-distribution of boron and the analysis of radiobiological effects; physicists and physicians for dosimetry, preparation of treatment plans and patient management.

This ambitious project could be realized putting in place a strong and multidisciplinary collaboration, involving the main academic and research institutes that have always been cooperating with CNAO since the beginning. The BNCT@CNAO Project will be focussing on integrating and expanding the current knowledge in BNCT in order to achieve the best possible beam from an accelerator based neutron source, possibly together with a novel boron molecule carrier. The four institutions constitute the core group of the pre-clinical and clinical research effort, but they are since the beginning open to new world-wide collaborations aiming to power the efficacy of this treatment modality.

Radiobiological research, dosimetric models and treatment planning simulations, boron measurements and clinical dosimetry will be duly pursued in order to verify the higher ability to selectively destroy neoplastic cells, sparing healthy ones, provided that the carrier capacity of "recognizing" neoplastic cells is very enhanced. In parallel, patients selection criteria, eligible pathologies and treatment protocols will be investigated. Research perspectives include also the development of new drugs for the administration of Boron-10.

# Development of novel compact activation-based neutron spectrometer for BNCT


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**Abstract text:**

In recent years, due to the introduction of accelerator based BNCT facilities, the interest of the medical and scientific communities for BNCT dramatically increased.
As far as the beam diagnostics is concerned, the availability of spectrometry techniques adapted to BNCT is recommended for QA purposes as well as for of comparing different facilities and accelerator types.
In the framework of the INFN project ENTER_BNCT an activation spectrometer with isotropic response, called NCT-ACS (NCT-Activation-Compact-Spectrometer), was developed.

The spectrometer consists in activation foils in cubic geometry, where every face is a sandwich of foils of different elements (In, Au, Mn, Cu, Na, Cl, V, Ti) with capture resonances ranging from the thermal to the epithermal region. Moderating shells of polyethylene can be eventually added to extend the sensitivity range, so that the response interval ranges from thermal up to 100 keV. Its response is isotropic. The energy distribution will be derived from unfolding the activation data.

Extensive simulations using the MCNP6 code for the choice of the device geometry and its materials composition have been performed. The most relevant results are shown in the contribution.

This communication presents also the preliminary measurements made with NCT-ACS at the LINAC-based neutron source of INFN-Torino/Università di Torino. These results can be considered as a first proof of the novel concept capability.


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# Application of gamma spectroscopy to alpha particles dosimetry in BNCT

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Radiotherapy using α particles is becoming an object of increasing interest. It is extremely important to monitor dose deposited by the particles in tissues and therefore – the dose rate during the therapy. Measurements of α particles delivered to the body in the form of radiopharmaceutical or boron carrier as a source of alpha particles in the reaction with neutrons can be only carried out indirectly. Difficulties in the direct *in vivo* measurements of α particles released in tissue are associated with their small ranges, on the order of μm. It is therefore necessary to develop appropriate measurement methods. A detection of low-energy γ photons accompanying the emission of these particles can be a suitable method [1].
Alpha particles are emitted, among the others, in the $^{10}B(n,\alpha)^{7}Li$ reaction. The reaction is the basis of boron neutron capture therapy (BNCT). In this reaction, with 94 % probability, γ photons with an energy of ca. 478 keV are emitted [2][3].
It will be presented a current state of research to develop a method for the determination of the dose deposited in a tissue-like medium by α particles derived from the $^{10}B(n,\alpha)^{7}Li$ reaction. The proposed method is based on the real time measurement of γ photons produced in the same reaction. A special experimental set up was designed and built to achieve this aim. Measurements were carried out in the field of neutrons emitted from Pu-Be source. Gamma photons were detected by a gamma ray spectrometer based on CdZnTe detector. PADC nuclear track detectors were used to determine the fluence of alpha particles at a given position inside of the boron solution filled phantom irradiated by neutrons. In addition, the simulations based on the MCNP code were performed in order to determine the efficiency of the $^{10}B(n,\alpha)^{7}Li$ reactions measurements.
The experimental set up built for the purpose of the measurements carried out in the MARIA Reactor at the National Center for Nuclear Research in Świerk, Poland, will be discussed. The conclusions drawn on the basis of measurement data obtained in this part of the research will also be presented.

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# A single moderator neutron spectrometer for neutron spectrometry in BNCT

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**Abstract text:**

According to recent recommendations of the IAEA BNCT expert group [1], a need exists for spectrometry-based instruments in the characterisation of therapeutic neutron beams.

In the framework of the INFN project ENTER_BNCT, a single moderator neutron spectrometer with directional response was conceived and prototyped.

The device, called NCT-WES (Neutron Capture Therapy - Wide Energy Spectrometer), condenses the functionality of a Bonner Sphere spectrometer in a cylinder embedding six thermal neutron detectors in previously optimized positions. NCT-WES is a polyethylene cylinder with 36 cm diameter and 41.5 cm height. To achieve a sharply directional response, the sensitive part is shielded with a thick barrier made of polyethylene and borated rubber, except in the collimator direction. Semiconductor-based thermal neutron detectors are used as internal thermal neutron detectors. The design was oriented to emphasize the spectrometric capability in the epithermal range and to achieve a limited weight (about 40 kg).

This communication describes NCT-WES, its response, construction, internal detectors, and the experimental tests conducted to date.

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# A new detector concept based on the Prompt Gamma Radiation Analysis for *in vivo* boron monitoring in BNCT

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The problem of born concentration monitoring during the BNCT therapy is one of the main challenges of this type of radiotherapy and is directly related to the nature of the interaction of neutrons with matter [1,2]. Among the available *in vivo* methods of boron monitoring Positron Emission Tomography (PET) seems to be very promising but it requires a new boron carrier with a $\beta^+$emitter which are not yet clinically available [3,4]. An alternative solution may be the Prompt Gamma Radiation Analysis (PGRA) based on the secondary radiation emitted in the interaction of neutrons with the patients tissues. It may provide not only the boron concentration determination but could contribute to the measurement of neutron flux and dose distribution estimation. This method requires however, compact gamma radiation detection systems able to sustain high counting rates and characterized by very good energy resolution. In this contribution we present the state-of-the-art solutions for monitoring in BNCT based on PGRA. Moreover, we describe a new concept of such system based on position sensitive scintillator detectors equipped with an anti-Compton shield which may provide determination of the gamma quantum direction supported with modern artificial intelligence algorithms.

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# A new model for characterizing the growth of spheroid

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Boron Neutron Capture Therapy (BNCT) is a two-step treatment aimed at curing or improving the life of patients who have been diagnosed with cancer. Nowadays, development of the new boron carriers, which are more efficient in delivery to the tumor cells, are one of the most promising improvements to BNCT technique. One of the requirements of the correct assessment of new carriers is the elaboration of an appropriate model that will allow to characterize the effect of irradiation on tumor growth [1].

One of the well-established experimental model used in research of the new cancer treatments and therapies are spheroids, three-dimensional cell culture that mimics the structure, microenvironment and cells signaling present in solid tumors [2-3]. The existing models of spheroid development do not allow for the correct characterization of the cells consisting of the spheroid. Therefore, we propose a new spheroid growth model which allows for a deeper insight into the structure of the spheroid, composed of three types of cells – proliferating, quiescent and dead [4-5]. We present a theoretical model which allows to accurately approximate the growth of the spheroid, additionally supported by results obtained from simulations. By the presented model, it is possible to separate growth of proliferating and non-proliferating cells, which may be substantial in forming an appropriate laboratory protocol to evaluate the effects of neutron irradiation. In addition, the simulation data allow not only to confirm the prediction from the model used, but also to check how cells in a given state are distributed inside the spheroids. In particular, the model allows an additional estimation of the fraction of dead cells indirectly, only from the growth curve. Therefore, the presented model can potentially provide more information than the standard approach in such studies of the growth dynamics of tumors – the Gompertz curve.

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# The MUNES accelerator-based thermal neutron source: microdosimetric characterization

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**Purpose:** The MUNES project (MUltidisciplinary NEutron Source) aims at the development of an intense accelerator-based thermal neutron source, suitable for Boron Neutron Capture Therapy (BNCT) [1]. A prototype of the final setup was developed and is installed at the CN accelerator of Legnaro National Laboratories. This work aims at the microdosimetric characterization of the resulting radiation field, in terms of spectral dose components and microdosimetric RBE, in view of BNCT applications of accelerator-based thermal neutron sources.

**Materials and Methods:** The source is based on a 5-MeV proton beam hitting a beryllium target. The emerging fast neutrons are moderated to the thermal range by a beam shaping assembly (BSA) made of a Polytetrafluoroethylene (PTFE) tank filled with heavy water and surrounded by graphite blocks. Microdosimetric measurements were performed with a cylindrical avalanche-confinement Tissue Equivalent Proportional Counter (TEPC) with interchangeable cathode walls [2], placed in front of the bismuth beam port. The sensitive volume is a cavity 13 mm in diameter and height, filled with pure propane to simulate a tissue-equivalent site size of 1 µm. Two cathode walls were used for the measurements, one doped with 100 ppm of boron and one undoped. A proper subtraction procedure [3] was applied on the measured spectra to determine the relative dose contributions due to photons, proton and ion recoils from neutron interactions on the wall materials, and BNC reaction products. The microdosimetric RBE was also calculated by means of a biological weighting function, and used to calculate the RBE-weighted biological dose [4].

**Results:** The comparison of microdosimetric spectra shows that the physical dose enhancement induced by boron doping is about 80%. The biological dose is however enhanced by a factor of 3, given the high RBE of BNC reaction products. In the presence of boron, BNC events contribute to 84% of the neutron biological dose, and 68% of the total. In the absence of boron, the physical dose is dominated by gamma events, with neutron-induced processes contributing only to 15%. However, given their higher RBE, their contribution to the biological dose increases to 39%. These values are in reasonable agreement with literature results, and confirm the suitability of this accelerator-based thermal neutron source for BNCT applications.

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# Development of an optically stimulated luminescence material with reduced ionization quenching for proton therapy dosimetry: $MgB_4O_7$:Ce,Li

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Luminescence quenching due to high ionization density remains one of the most important corrections for applications of luminescence dosimetry in ion beam therapy measurements. Nevertheless, the development of new optically stimulated luminescence (OSL) materials with increased linearity range and high dose saturation opens the possibility of reducing such corrections.

The objective of this work is to demonstrate a new OSL dosimetric material specifically developed with improved response for proton therapy dosimetry, i.e. with less reduction in luminescence detector efficiency (ionization quenching) with the particle linear energy transfer (LET).

$MgB_4O_7$:Ce,Li was synthesized and its luminescence efficiency as a function of LET was characterized for dose-averaged LET values in water from 0.73 keV/µm up to 74.9 keV/µm using proton, He-4, C-12 and O-16 ion beams at the Heidelberg Ion Beam Therapy Center (Heidelberg, Germany). Commercial $Al_2O_3$:C OSLDs were also used for comparison. Monte Carlo simulations were used to predict the material's response in a spread-out Bragg peak (SOBP), suggesting the $MgB_4O_7$:Ce,Li would offer minimum quenching in proton beams of energies relevant to radiotherapy. This prediction was confirmed by irradiations of $MgB_4O_7$:Ce,Li and $Al_2O_3$:C in pristine and SOBP proton beams performed at the Center for Proton Therapy at the Paul Scherrer Institute.

The results confirm the hypothesis that new OSL materials optimized for proton dosimetry can be developed using their dose response to low LET radiation as a guide.

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# Measurements of neutron and gamma dose to the fetus for pregnant patients undergoing brain tumors proton radiotherapy with modulated scanning beam

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Due to the high radiosensitivity of fetus, pregnant patients are referred to radiotherapy only when there are strong clinical indications. Despite the fact that proton radiotherapy (PT) may reduce the dose receiving by the fetus comparing to conventional photon therapy [1], the embryo exposure to radiation should be estimated very carefully. However, the calculations accuracy of out-of-field doses with commercially available Treatment Planning Systems is not sufficient for this purpose. This study presents a methodology to perform fetal dose measurements, including the in-vivo neutron dose estimation.
The geometry of 25-week pregnant women voxelized (CT) phantom [2] was set-up in treatment room using Alderson Rando anthropomorphic phantom (head and chest) and RW3 plates (abdomen and pelvis). Four different irradiation plans were prepared in order to irradiate the target (volume of 22.3 $cm^3$) located in brain with proton beam (PBS technique). The out of field dose measurements at the location of the fetus were performed using WENDI-2 detector (neutrons) as well as FHT192 probe (gamma) for all plans. The ambient dose equivalent, H*(10) measured by WENDI-2, was converted to organ dose equivalent (H) using simulated spectra from MNCP 6.2 simulations and applying corrections according to the WENDI-2 neutron energy response function. Additionally, the bubble detectors (BD-PND) were placed at three different locations in the phantom to measure the fetus neutron dose equivalent and explore their suitability for in-vivo measurements.
Depending on the treatment plan geometry, the neutron dose was measured to be between 36 and 59% of total dose (gamma + neutron) to the fetus, which confirms the importance of neutron dose estimation for accurate fetus dosimetry in PT.
Comparing to WENDI-2, the neutron dose equivalent underestimation at the level of 30-60% for bubble detectors was observed for the measurements at the fetus location. These discrepancies can result from different size of both detectors, which implicates different geometry of measurements, while the dose gradient over the uterus can be significant. Also the uncertainty of the methods needs to be considered (both around 20%). The dose measured with bubble detectors located at patient chest was within its dynamic range for a single fraction, which makes it suitable for in-vivo dosimetry method. The determination of conversion factors for such in-vivo measurement to dose measured at embryo location with WENDI-2 or bubble detector in the phantom allows to estimate the fetal dose for each fraction.

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# Characterization of LiF:Mg,Ti for dosimetry in ultra-high dose rate proton beam

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This work aims at characterizing LiF:Mg,Ti (TLD-100) detectors for dosimetry in proton beams delivered with ultra-high dose rates (UHDR). On the one side, the study focuses on the investigation of possible dose rate effects in LiF:Mg,Ti material. On the other, it addresses the challenge of dosimetry in narrow proton beams, where the signal averaging over the detector surface causes a dose under-response.

Small (1.0 x 1.0 x 1.0) $mm^3$ TLD-100 cubes were annealed and packaged in matrices consisting of 25 TLDs in a 5 x 5 grid. The TLDs were read out using a LexsygSmart automated reader, where the signal of each TLD was normalized using the built-in beta source to reduce the inter-sample variability. The center of each matrix was irradiated with dose rates (1-1400) Gy/s, and the narrow pencil beam was subsequently reconstructed using a 2D Gaussian fit to the doses derived from the 5x5 TLD grid. The dose delivery was monitored with a Faraday Cup, whilst radiochromic films were placed in front of the TLD-100 packages to verify the peak dose and spot size estimated with the TLD matrices.

The peak dose derived from the Gaussian fit to the TLD grid agrees with the dose measured with the radiochromic films within the experimental uncertainties. For the dose rates under investigation, no dose rate effects are observed in LiF:Mg,Ti detectors.

The matrix of LiF:Mg,Ti TLDs combined with the beam reconstruction can thus be applied for the verification of the dose and the beam profile of narrow proton beams. The dose rate independence of LiF:Mg,Ti makes them suitable for dosimetry in UHDR proton beams.

# Neutron spectrometry for medical and industrial accelerators

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Neutron is one of the most produced secondary particle around medical and industrial accelerators, both by nuclear inelastic (protons, ions) and photo-nuclear ($\gamma$,n) reactions. In addition to neutron activation issue, neutrons also require a particular attention due to their long range and their high relative biological effectiveness (RBE) [1]. A precise knowledge of the produced neutron field for a given medical or industrial application is thus important to improve the radiation protection of workers and population in these installations. Monte Carlo (MC) simulation and analytical codes for radiation-matter interactions are considered to be powerful and practical tools for this purpose. However, there is still today a lack of nuclear data necessary to validate models of neutron production used in theses codes.

The reference method for neutron spectrometry is the Bonner Spheres System (BSS), which allows to reconstruct the neutron spectrum over a wide energy range from meV to GeV. However this detection system is hardly compatible with the constraints of most medical and industrial environments, especially because of its large size and its saturation at high fluxes [2]. In order to facilitate the acquisition of nuclear data in medical or industrial installations, it seems interesting to develop more dedicated neutron spectrometry system. The IPHC laboratory (Strasbourg, France), has been working for a few years on the conception of such an innovative neutron detector [3]. This spectrometer is a recoil proton telescope (RPT) designed to operate within 5 to 30 MeV energy range. The instrument is made of a thin polyethylen converter (typically 50 µm) to generate recoil protons, 3 pixelated CMOS sensors specially designed for high rate proton tracking, and a silicium diode to measure the proton energy. The detector is extremely compact (10x10x8 $cm^3$) to be easily used in various medical and industrial irradiation facilities. Its high rate detection due to CMOS performances, allow it to work in high fluxes environment, up to $10^9$ n/s/$cm^2$. The neutron spectrum is reconstructed in real time, without complex and time consuming unfolding process.

The RPT performances have been evaluated both with protons (Cyrcé cyclotron – IPHC - France) and neutrons beams (AMANDE facility – IRSN – France), for the proton tracking and the neutron spectrum reconstruction. An *in situ* calibration method for the diode, based on inelastic reactions in silicium has been successfully tested in mono energetic neutron field. The detector will be tested at the end 2021 in a proton therapy center (Center Antoine Lacassagne - France) and in an industrial irradiation facility (Aérial Feerix – France). Instrumentation, performance evaluation and first data measured at medical and industrial facilities will be presented in this work.

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# 2D OSL dosimetry based on $LiMgPO_4$ foils - facilitating the high-resolution proton radiotherapy

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Modern radiotherapy (RT) techniques, such as proton radiotherapy, have enhanced the geometrical accuracy and complexity of the delivered dose distributions. Thus, spatial two-dimensional (2D) or even three-dimensional (3D) resolved measurements are required to properly validate complex treatment plans. These also imply the development of new kinds of dosimetry techniques and materials. One of the new and promising approaches to the 2D dosimetry,, based on technology consisting of flat and flexible sheets made of a polymer, with the embedded optically-stimulated luminescence (OSL) material in the form of powder ($LiMgPO_4$, LMP), and a self-developed optical imaging setup, consisting of an illuminating light source (using blue LED's) and high-sensitive CCD camera [1,2]. The newly developed 2DOSL system has been tested during the verification of the real 3D proton depth dose distribution of the eyeball cancer treatment, using a dedicated patient collimator and 60 MeV modulated proton beam [3]. Although the reconstructed 3D proton depth dose distribution matches very well with the clinical treatment plan, the data showed a very-well know effect of lower luminescent efficiency of the LMP material to protons, similar to what has been observed previously for the other OSL materials, and being related to the ionization density of the heavy charged particles [4].
The resulting dependency of the LMP OSL luminescence response on proton energy, suggesting the need to apply efficiency corrections, especially that to our knowledge, such data are not available at present for LMP material. Therefore in the present work, the prototype LMP based foils were tested with monoenergetic uniform proton fields for various energies constituting the so-called Spread-out-Bragg-Peak (SOBP). The measurements are realized exploiting the 60 MeV proton beam of the AIC-144 cyclotron in the Proton Eye Radiotherapy Facility at IFJ PAN. The influence of proton energy on the relative OSL luminescence efficiency and other OSL characteristics of the studied LMP foils will be presented.

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# Study of the p-$^{11}$B nuclear reaction for the enhancement of proton therapy through a silicon telescope at CATANA facility


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Given the first experimental proof of Proton Boron Capture Therapy for the enhancement of proton therapy [1], based on the p+$^{11}$B→3α nuclear reaction, the NEPTUNE project was established with the main goal of investigating both the physical and radiobiological phenomena that could justify an increase of the proton-induced cytogenetic effects in cells irradiated in presence of a boron-based agent.
In this work, a double-stage silicon telescope coupled to different boron targets was irradiated at the CATANA proton therapy facility for studying the proton boron reaction by discriminating secondary particles from primary protons.
The structure of the silicon telescope consists of a ΔE stage and a residual-energy E stage about 1.9 μm and 500 μm in thickness, respectively. The E and ΔE signals were collected separately in two different electronic chains, in order to discriminate the type of the impinging particle through the so-called “scatter-plot”.
Two different boron targets were designed and developed by depositing boric acid ($H_3BO_2$), both enriched up to over 99% in $^{11}$B or $^{10}$B, on a 50 μm thick PolyMethilMetacrylate substrate: the $^{11}$B enriched target allowed to study the effect of the p+$^{11}$B→3α nuclear reaction directly triggered by primary protons, while the $^{10}$B target allowed to evaluate the contribution of secondary particles from boron neutron capture reaction triggered by secondary thermal neutrons.
Measurements were performed at 30.3 mm in water depth across the Spread Out of Bragg Peak of the 62 MeV proton beam at CATANA therapeutic facility: this depth was selected for maximizing the alpha particle production, according to the reaction cross section (the main resonance is at 675 keV).
By exploiting the measured scatter-plots, it was possible to discriminate events due to alpha particles produced in the boron target from primary protons traversing the target itself. This discrimination allowed to evaluate the energy distribution of the alpha particles and their imparted energy in the ΔE stage [2]. Even if a reliable quantitative study of the alpha production rate has not been achieved yet, this work demonstrates that low energy and, therefore, high-LET particles from the reaction triggered by low energy protons on the $^{11}$B can be measured. In the next future, dedicated experiments with low-energy proton beams and thin boron targets are foreseen for a quantitative analysis on the production rate.

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# Evaluation of the LET measurement capabilities of various Optically Stimulated Luminescence detectors

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The aim of this work is to improve LET measurements in proton dosimetry by evaluating the performance of different optically stimulated luminescence (OSL) materials, including $Al_2O_3$:C, $Al_2O_3$:C,Mg, and the newly developed $MgB_4O_7$:Ce,Li. LET measurements are important to correct the non-linear response (ionization quenching) of the OSL detector for a dose measurement as well as to provide information about the biological dose in proton therapy.

The OSL detectors were read out in a commercially available reader in a pulsed mode and subject to a reference irradiation with a built-in beta source. The pulsed stimulation mode enables a distinction between the fast and slow emission bands, whereas the reference irradiation serves to correct each OSL for material and size differences.

It is demonstrated that the ratio of the slow and fast emission bands in the $Al_2O_3$ based OSLs correlate with the LET, in particular for $Al_2O_3$:C,Mg, which enables LET calibrations and measurements. The $MgB_4O_7$:Ce,Li OSLs are, on the other hand, almost quenching-free which makes the material suitable for dose measurements in proton beams and as reference for the quenching $Al_2O_3$ based OSLs. The individual OSL correction factors derived from the reference irradiations following readout are demonstrated to improve the reproducibility of both LET and dose measurements.

The pulsed-mode readout of the OSL materials, along with the reader with its built-in source, allows for improved simultaneous dose and LET measurements. Furthering previous studies, it is shown how the technique can be used to measure dose and LETs across the range relevant to proton therapy.

# Neutron dose assessment in laser generated ultra short pulsed fields

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ELI Beamlines is one of the pillars of the ELI ERIC (Extreme Light Infrastructure European Research Infrastructure Consortium), the European project aiming at building the next generation of high power lasers for fundamental research and industrial applications.

Several high power lasers are hosted by the ELI Beamlines facility. Among these lasers is the 1 PW HAPLS (High-repetition-rate Advanced Petawatt Laser System) laser, which has started providing its beam to experimental stations in 2021, with a ramp up in power, while commissioning experiments were being performed. Even at power lower than the nominal one, when interacting with a target, the laser can generate mixed ionizing radiation fields of unique nature.

Detecting the neutrons generated during the commissioning experiments has been a challenging endeavour, since certain difficulties were faced. First, the experimental conditions were indeed frequently altered during the commissioning phase (such as laser beam parameters, experimental geometry, or target type). Next, the extremely short duration of the ionizing radiation pulse generated by the laser (~$10^{-14}$ s) made the correct interpretation of the data provided by the detectors designed and calibrated in (more or less) standard fields complicated. Another factor that had to be considered, was the electromagnetic pulse associated with a high power laser pulse interaction with targets.

In this contribution, we will describe the experiments performed and how the problem of the detection of the ionizing radiation was addressed. Preliminary results of radiation measurements will be presented, together with the lessons learned and how these will be exploited in future experiments.

# Radiation protection measurements during the commissioning of the third-generation n_TOF neutron spallation target at CERN

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The neutron Time-Of-Flight (n_TOF) facility, operational at CERN since 2000, is a pulsed neutron spallation source producing neutrons between a few meV and several GeV for two experimental stations located 185 m horizontally from the target and 20 m above the target. The neutron production target, made of pure lead, is impacted by the 20 GeV/c proton beam of the Proton Synchrotron. Experiments performed at n_TOF find applications in nuclear astrophysics, nuclear technology and medical research. In 2018, the second-generation target reached the end of its lifetime, and the facility underwent a major upgrade during CERN's Long Shutdown 2 (2019-2021), which included the installation of the new third-generation spallation target [1]. The previous targets were based on water-cooled massive lead blocks, whereas the new target is cooled by nitrogen gas slightly above atmospheric pressure to avoid erosion-corrosion and contamination of the cooling circuit with radioactive lead spallation products. A radiation protection study, performed with the FLUKA Monte Carlo code [2], contributed to the optimisation of the target design and to assess the safe operation of the facility. In 2021, during the commissioning of the new target, the simulation results were benchmarked against and complemented by dedicated radiation measurements, which are described in this work:

- Monitoring of stray radiation (neutrons and photons) in accessible areas by using fixed and mobile instrumentation (high-pressure ionisation chambers and the LUPIN extended range rem-counter [3], specifically conceived to measure pulsed neutrons) to ensure the compliance with the radiological area classification.
- Monitoring of the nitrogen activation and contamination levels by means of a flow-through differential ionization chamber, for which an in-field calibration factor was calculated combining FLUKA simulations and experimental measurements.
- Monitoring of the activation levels of moderator water by means of $\gamma$-spectrometry and liquid scintillation measurements.
- Monitoring of residual dose rates along the n_TOF beam transfer line to identify locations of possible beam losses.
- Activation studies for zoning purposes by placing samples of materials typically used in particle accelerators (aluminium, copper, stainless steel).

This contribution will report the above-mentioned technical aspects and provide perspective for operation in the next years.

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**Progress in the assessment of neutron environments associated with fusion energy technologies**

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**Abstract**

The most promising technologies for near-term fusion power generation are likely to be fuelled using deuterium and tritium, which releases 17.6 MeV per fusion reaction, 80% of which is carried by a fast neutron. Fusion energy will be characteristically neutron-rich compared to other nuclear processes such as fission and those neutrons will necessarily be mostly utilised for tritium fuel and high-grade heat production in a lithium containing blanket surrounding the fusion source. Unused neutrons will require effective nuclear shielding designs, elaborated through detailed neutronics and experimental underpinning to demonstrate minimisation of occupational dose associated with operations and maintenance, and for economic factors in protecting plant and equipment. Evaluations of the nuclear safety for fusion power plant concept designs requires detailed knowledge of the plant construction, materials, operational and environmental conditions to provide input to predictive, validated nuclear simulation tools. Additionally, a system of robust, traceably calibrated neutron dosimetry systems that are deployable in the plant are needed for accurate measurement of workplace environments.

UKAEA develop and operate a wide range of neutron diagnostics and modern spectrometry systems that underpin experimental activities in fusion research at Culham. We discuss capabilities and recent developments in these areas, covering neutron diagnostics development and optimisation activities, and the latest neutronics modelling tools that are used to predict the nuclear fields both during and following tokamak operations.

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# Occupational Exposure in CANDU Nuclear Power Plant - Individual Dosimetry Program at CNE Cernavoda NPP

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CNE Cernavoda NPP has two CANDU 600 reactors in commercial operation, first since December 1996 and the second one since November 2007.

For a CANDU 6 type reactor the major contributor (90%) to the external dose is gamma radiation. The major contributor to the internal dose of professionally exposed workers is the tritiated heavy water (DTO) – 16 ÷ 30% of the total effective dose.

The main purpose of design and implementation of a “Monitoring, Evaluation and Recording of Individual Doses Program” (Individual Dosimetry Program) is to measure, assign and record all the significant radiation doses ($H_p(10)$, $H_p(0.07)$ and $E_{50}$) received by an individual during activities performed at CNE Cernavoda NPP and ensure that all the exposure are kept ALARA.

We provide dosimetry services for measurement, evaluation and recording of all significant ionizing radiation doses received by workers and visitors at CNE Cernavoda, due to the presence of radioactive sources or working with nuclear facilities.

For all the persons entering radiological controlled areas (CNE Cernavoda employees, short-term atomic radiation workers, contractors and visitors) Health Physics Department provides individual (external and internal) dosimetric surveillance.

Individual dose monitoring is provided by a licensed dosimetry service, approved by the Romanian regulatory body, National Commission for Nuclear Activities Control (CNCAN), at CNE Cernavoda.

Radiation Protection Program for Cernavoda NPP is designed insomuch as legal dose limits, established by our national regulatory body, for atomic radiation worker and for public, will not be exceeded.

CNE Cernavoda Dose Monitoring, Evaluation and Recording program is based on the latest ICRP recommendations and also, on the EU requirements and national laws and regulations.

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# Eye Lens Dosimetry in Canadian CANDU Nuclear Power Plants Based on Operational Dosimetric Quantities $H_p(10)$ and $H_p(0.07)$

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In this paper, we present a methodology for performing eye lens dosimetry in CANDU nuclear power plants using an existing and highly accurate Harshaw 4-element TLD-700 dosemeter. This dosemeter, which has been specially designed for Ontario Power Generation (OPG) and Bruce Power (BP), measures the deep and shallow personal dose equivalent quantities $H_p(10)$ and $H_p(0.07)$, respectively. Using these measured personal dose equivalent quantities and applying a beta-ray strength scaling factor to the $H_p(0.07)$ measurement in particular, we have developed an algorithm that can be used to calculate the dose to the lens of the eye in mixed beta-gamma fields. This scaling factor has been developed and is primarily based on results obtained from extensive collaborative study, performed by Ontario Power Generation (OPG), Bruce Power (BP) and McMaster University, through Candu Owners Group (COG) support [1], [2]. Furthermore, scaling factor F, also includes effects of protective glass eyewear and results from Whole body dosimetry intercomparison exercises. The algorithm to calculate eye lens dose at CANDU power plants has been developed, based on this scaling factor and operational dosimetric quantities $H_p(10)$ and $H_p(0.07)$.

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# Dose assessment in a real case of internal contamination with $^{177}$Lu through a wound.

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Lutetium-177 is a β− radionuclide used typically for theranostics having a half-life of 6.7 days and γ-photons emissions which are used for diagnostic evaluation and patient therapy. A worker was internally contaminated with $^{177}$Lu through a wound on 9 March 2021 in a radioactive facility as consequence of a puncture in the middle finger of the left hand, with a needle that had been in contact with the inside of the closed stock vial, with 2 ml of Lutetium Chloride ($LuCl_3$) and with an activity of 162.5 GBq of $^{177}$Lu.

Successive controls of contamination levels have been carried out over the area of the puncture using a contamination monitor. Before decontamination the measurement carried out on the gloved hand gave an “overflow” with a reading above the equipment range (> 50 kcps). The operator then removed the gloves and after a preliminary decontamination, he measured again in contact with the affected area, resulting in a count rate of 34 kcps.

In vivo measurements of gamma emitters in the total body of the contaminated worker were carried out at CIEMAT Internal Dosimetry facilities using a big NaI(Tl) detector inside the shielded room of the Whole Body Counter (WBC) Laboratory, in a reclined chair counting geometry. The contaminated hand was shielded with a wrapping lead material. The main gamma emissions used for the activity estimation of $^{177}$Lu in total body were 208 keV (11.0%) and 113 keV (6.4%).

The biokinetic model of the Lutetium for injection was used for internal dose evaluation. The total body retention fractions on days 3 and 7 after the incident and the dose coefficient of 2.4E-10 SvBq$^{-1}$ were available from the Data Viewer (electronic annex) of ICRP Publication 141. A committed effective dose E(50) of 0.23 mSv was calculated based on activity values of 563.0 ± 93.1 kBq (2σ) and 359.0 ± 59.5 kBq (2σ) of $^{177}$Lu in total body obtained on 12/03/2021 and 16/03/2021 respectively. The activity of the $^{177}$Lu incorporated into the blood stream through the wound puncture was estimated 958 kBq. Both total body activities in 2 different days confirmed well this assessment.

Furthermore, two urine spot samples were collected and analysed by gamma spectrometry at the CIEMAT in vitro bioassay Laboratory. The result of the activity in the second sample collected on 16/03/2021 resulting in a good agreement with the uptake of $^{177}$Lu estimated from the two in vivo monitoring data.

In addition, measurements over the puncture area were performed at CIEMAT WBC, using 2 BE Ge detectors at a distance of 15 cm from the damaged finger and inside the shielded room, considering a counting geometry of point source. The results of activity estimates were 269.6 ± 43.5 kBq (2σ) and 125.9 ± 20.4 kBq (2σ) of $^{177}Lu$ on 12/03/2021 and 16/03/2021 respectively. ACPRO carried out the skin dosimetry calculations for the worker: the maximum equivalent dose averaged over 1 $cm^2$ of skin has been of the order of 228 mSv. The equivalent dose in hand has been estimated 8.8 mSv.

# Characterization of a new detection system on the MIRCOM facility

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Over the course of the last years, ion microbeams have become important in several fields. In radiation biology, for example, studies aim at identifying, understanding, and preventing the side effects resulting from the use of ionizing radiation, for therapeutic purposes in radiotherapy, and more particularly in hadrontherapy. To carry out this research, the Institute for Radiological protection and Nuclear Safety (IRSN) is equipped since 2018 with the MIRCOM facility to perform targeted micro-irradiation of living biological samples such as cells or small multicellular organisms with a given number of charged particles. It is based on a 2 MV Tandetron™ accelerator manufactured by HVEE [1] that can produce protons up to 4 MeV, alpha particles up to 6 MeV and heavier ions such as carbon and oxygen up to 8 to 10 MeV. Before reaching the biological samples, the ion beam is focused, down to less than 1 µm in diameter in vacuum, and then extracted in air by a silicon nitride vacuum window ($Si_3N_4$). An epi-fluorescence microscope is located in front of the microbeam extraction and is used to locate and select the areas of interest of a sample, and to visualize in real time the early effects of the irradiation using time-lapse imaging.

Between each irradiation, the ion beam is monitored by a PIPS (Passivated Implanted Planar Silicon) detector, mounted on the microscope’s objective wheel facing the microbeam. Its active surface is thick enough for the ions to deposit all their energy so it is considered to have a 100% detection efficiency. However, the ion range at the energies available on MIRCOM is too low for them to reach the detector when a biological sample is in place. The PIPS detector is thus not adapted to use to monitor the ion beam during an irradiation. Consequently, to improve the accuracy on the number of ions delivered on target, especially at low ion number, a detection system located upstream the biological sample has to be used.

In this general context, developments have been performed on MIRCOM. Indeed, when ions pass through the extraction window, secondary electrons are emitted. These electrons can then be collected by a Channeltron to count the number of ions passing through the window. This system is currently used on the microbeam line of GSI (Germany) [2]. To increase the production of secondary electron, and thus the detection efficiency, the windows have to be coated, on the vacuum side, with electron emitting materials, such as gold and caesium iodine.

This presentation will describe measurements performed using the Channeltron with different thicknesses of Au and CsI and compared with the ones simultaneously carried out with the PIPS detector. A detection efficiency of 100% for alpha particles was reached but an optimization of the coating thickness is still being assessed to limit, as much as possible, the induced energy degradation and beam straggling. Similar work will be carried out for the carbon and oxygen ions.

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# Big data and machine learning: can this be used in personal dosimetry?

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Individual monitoring of workers is mostly performed by using personal dosemeters, both active and passive ones. Although there has been a consistent improvement in dosemeters technology, large uncertainties still persist in personal dosimetry. Also, many practical problems exist, with many dosemeters getting lost and the reluctance of workers to wear one or more dosemeters.

The fast progress in artificial intelligence (AI) opens up new possibilities and perspectives for radiation protection in general. In this contribution, we try to explore if and how big data and machine learning could potentially also be useful for personal dosimetry.

The PODIUM project (Personal On Line Dosimetry Using Computational Methods) showed that by using powerful computational methods based on simulations and motion tracking, individual doses can be calculated with the help of motion tracking of the workers. Speeding up the simulations can be done by introducing machine learning (ML) algorithms that prioritize simulations based on a priori calculated results. The effect of important influence parameters on the worker doses can be taken into account to optimize simulation times. If sufficiently large databases are available with precalculated or measured doses or dose rates, it could even be possible to use these results to estimate the doses based on worker movements.

Such computational and ML approaches are useful for fields where the doses to the workers can be significant. But the large majority of workers only occasionally have doses above the detection threshold.

Typically, in personal dosimetry, a lot of prior information is available. By collecting the history of doses of workers in a certain work environment, an extensive database of a priori information can be generated. This dose database can then be combined with operational information of the worker (e.g. classification of work area, number of entrances to a specific controlled area, workplace monitoring results, effective working time, performed activities, …) to generate a comprehensive dataset of exposure parameters for each worker.

It is possible to apply machine learning parameters on such “big data” sets to find correlations and set-up prediction models. Such “dose predictions” based on a combination of multiple parameters, can determine the probability of the worker receiving a significant dose during that activity or period. This dose probability can be used to optimize the dosimetry service. It can be envisaged that dosemeters with very low probability can be read out in low priority. If the estimated dose probability is high, extra information can be fed to the model (like more precise information of the location of the worker in the field based on camera images) to even predict the dose value itself.

In this contribution, we have applied this principle as a proof of concept on the doses from part of the SCK CEN staff. We have used the dose history from over 10 years at several locations at our research institute, and tried to set up sophisticated ML algorithms for predicting dose probabilities.

# Thermal effects on decolorization of the PVA-iodine complex containing silica nanoparticles

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While it was found several years ago that the complex of polyvinyl alcohol, iodide and silica nanoparticles (PAISiN) has a reversible radiochromic property [1], the effects of temperature on the stability of the radiation-induced red color were unclear. Thus, we investigated the thermal effects on decolorization of the PAISiN complex. Three samples each of the PAISiN complex were put in prismatic glass cells and were irradiated with 5, 10 and 20 Gy of Cs-137 γ-rays. The changes of the radiation-induced colors were observed at different temperatures: 20°C (room temperature), 40°C (in an oven) and 5.5°C (in a refrigerator). At 20°C, the sample irradiated with 20 Gy returned to the original white color from the dense red color in 3 days. At 40°C, the process of decolorization was remarkably accelerated and completed in 3 hours. On the other hand, no decolorization was seen at 5.5°C for up to 10 days. According to these findings, it is considered that the PAISiN complex has a practical feature as we can control its radiation-induced coloring status by adjusting temperature. More concretely, we could accelerate decolorization of the PAISiN complex by heating for reusing the same sample immediately, while we could maintain the radiation-induced coloring state by cooling for future analyses. Applications of the PAISiN complex for monitoring of possible high-dose exposures in medicine and industries are to be investigated.

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# Experimental doses in the laser processing laboratory

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The irradiation doses from the lasers with the high power and high repetition rate dose can accumulate over time and create a radiological hazard. When laser peak intensity at target surface is higher than $10^{13}$ W/cm$^2$ a laser-induced plasma is generated, which acts as X-ray source [1]. Plasma electrons are accelerated in driving laser field and interacts with materials generating bremsstrahlung and characteristic radiation. We present experimental results aimed at evaluation of maximized irradiation doses in the laser processing laboratory.

The laser radiation was focused into a spinning targets of different metals. Maximum laser intensity of $4.7\times10^{14}$ W/cm$^2$ was achieved. Two types of dosimeters were used for experimental dose measurements. Electronic semiconductor dosimeters were used for dose rate measurements. Thermoluminescence dosimeters (TLDs) were used to monitor integral doses at the different spots in the laboratory during whole experimental session. The same type of dosimeters were used for personal dosimetry. Amptek XR-100CR X-ray spectrometer with Si detector was used to measure spectra of X-ray.

Single pulse and double pulse regimes were investigated [2]. Introduction of a optimized pre-pulse before main pulse significantly increased the X-ray yield generated on the target, therefore this regime was used for experiments.

The dose rate at 35 cm distance from source measured using spectrometer was calculated. The results show that per 2080 hours (typical work year) the integral H*(10) dose at this distance would be 18.95 Sv and the integral H'(0.07) dose would be 2633 Sv, which is more than 4 orders higher than recommended dose for public users. TLDs gathered dose for 3 weeks (15 days in the lab), both H*(10) and H'(0.07) doses were found to be well above the regulatory limits. It was also found that 3 mm iron shielding was sufficient to attenuate the dose below recommended limits. From the spectral data we see that most of the energy is contained in the lower energy part of the spectrum, i. e. ordinary electronic dosimeters are not capable to measure significant part of the irradiation dose resulting from low energy X-ray photons.

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# Construction improvements of recombination chamber used for radiation protection

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Recombination chambers are used for radiological protection with success for many years. Lately the chamber was reconstructed to achieve equality in detection gamma and neutron radiation [1].
Main improvements was based on raising number of hydrogen atoms in composition of detector. Material of inner electrodes have been exchanged from polyethylene to Polipropylene. Such improvement give almost equal sensitivity of novel detector to gamma radiation and to neutrons in mixed radiation field. It results in enabling recombination chamber to be used as an H*(10)-meter. It is very convenient for environmental measurements and shorten time of measure without exchanging of detector.
Another improvement was done by constructing two sets of electrodes in one detector to which different voltage can be applied. Such step shorten further the time of measure and moreover enable users not to change voltage and by the same time give information about radiation quality.
Last but not least improvements corresponds to construction of self-monitoring detector which can be used in not stable radiation fields. The construction is basing on combining in one detector two separate parts with different gas filling and different voltage applied to each part.
Some construction ideas was planned and predicted by Monte Carlo calculations [2] and finally calibrated in routinely used gamma and neutron radiation sources i.e. Cs137 and Pu-Be.

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# Dosimeters intercomparison in ultra-short pulsed stray radiation field at SwissFEL

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Recently, several X-ray free electron laser facilities entered into operation, and, at the same time, laser-driven accelerators began to become viable as an appealing alternative to radiofrequency-driven machines in terms of acceleration gradient. Both technologies produce pulsed stray radiation fields, mainly high-energy photons, characterised by exceedingly high dose rates confined in ultra-short (ps-fs) pulse widths. In these conditions, the performances of the current personal dosimeters shall be verified and compared to ensure the accuracy of personal dose monitoring.
In this framework, the EURADOS WG11 initiated a dosimeters intercomparison program in 2017 with a measurement campaign at a pulsed 6 MeV electron LINAC at the Lausanne University Hospital (Lausanne, CH) [1]. This was followed by tests in the more challenging field conditions of SwissFEL (Paul Scherrer Institute, Villigen, CH), described in this work.
The SwissFEL electron beam has energies of 3-4 GeV, a 10-50 fs pulse width and a repetition rate up to 25 Hz, yielding dose rates in the order of 10 GSv/h within a single pulse. For the measurement campaign, the electron beam was steered to hit beam line components to produce bremsstrahlung. The measurements were performed with Ar/N-filled ionisation chambers (Nausicaa), passive thermoluminescence (TLD100, TLD600, TLD700), optically stimulated luminescence dosimeters ($Al_2O_3$:C, $Al_2O_3$:C Luxel, BeO), Direct Ion Storage (DIS-1) and electronic personal active dosimeters PM1300 (Polimaster, BY) on both sides of a PMMA ISO slab phantom.
Results from Nausicaa showed a considerable underestimation if compared with the passive dosimeters, which agree within the respective statistical uncertainties. Monte Carlo simulations performed with FLUKA [2] allowed investigating the dose contribution of each radiation component and studying the related response function of Nausicaa. Results shows that the Nausicaa underestimation could be attributed to its limited response to electrons and positrons, and not to the pulsed time structure of the radiation field.
The intercomparison program endorses its usefulness by highlighting the main limitations and challenges, settling dosimetry references in such extreme ionising radiation environments, and fostering additional research in ultra-short pulsed fields.

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# Radiation protection at ultrashort-pulsed lasers in materials processing

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Ultrashort-pulsed lasers with pulse durations in the sub-picosecond range are increasingly used in materials processing. Such short pulses with a high intensity allow accurate material removal without heating up the surrounding material. They are used, for example, for cutting Gorilla Glass for cell phone displays or for drilling injection nozzles for lower-emission engines.

In these processes, single laser pulses with very high intensities are fired at a workpiece, generating a plasma for material removal. The high-energy plasma electrons accelerated by laser-plasma interactions can emit X-rays with photon energies of more than 5 keV when the laser intensity exceeds $10^{13}$ W/cm$^2$ [1,2]. Due to this undesired emission of X-rays, ultrashort pulse laser machines are subject to the German Radiation Protection Act (StrlSchG). Several regulations and concepts for ensuring the radiation protection at these machines are under development.

In this contribution, the generation of X-rays and the radiation protection issue will be addressed, and first measurement results will be presented. As the emitted X-rays are pulsed and of very low photon energy, generated in a complex and unstable process, the radiation exposure measurements are challenging. Detailed measurements will be shown in a separate contribution.

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# OPTIMIZATION AND CHARACTERIZATION OF BI-DETECTOR COINCIDENCE BETA-RAY SPECTROMETRY SYSTEM

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We present characterization of a Si-plastic scintillator coincidence beta-ray spectrometer. With the recent recommendation to lower the dose limit for the lens of the eye by International Commission on Radiological Protection, beta-ray dosimetry is of great importance for nuclear industries, particularly during the maintenance periods. Since the beta-ray spectral data is most fundamental and vital information for beta-ray dosimetry, we have been developing a Si-Plastic scintillator coincidence beta-ray spectrometer, which aims to collect pure beta-ray spectral data by rejecting the gamma-ray detection events through coincidence. The pulse height and arrival time of each detector signal was processed by a compact digital system and was collected in list mode. With a recent upgrade in digital system, the spectrometer can cover the entire beta energy range of interest that is encountered in nuclear power plants. The responses of the spectrometer to beta and gamma were characterized by experiments and Monte Carlo simulations. For performance evaluation, spectral measurements were carried out using mixed beta-gamma fields with various beta and gamma count rates using $^{90}$Sr and $^{137}$Cs sources. For a fixed beta field, the coincidence spectrum was quite stable and consistent in most energy region with the increase of the gamma count rate while the count rate increased in the low energy region. Development of a realtime spectrum analysis method is currently underway.

# MEASUREMENTS IN PULSED NEUTRON FIELDS

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Dosimetry in pulsed and mixed radiation fields represents an important challenge in radiation measurements. This issue is explicitly addressed by the EURADOS Strategic Research Agenda (SRA) [1] that highlights the need of developing new detectors and associated electronics. In several accelerator technologies, bunches of particles are accelerated in a short time, producing intense radiation pulses spaced by a relative long time of beam off. An example are medical LINACs, in which electron pulses have a duration of few µs and a repetition rate from tens of Hz up to 1 kHz. Synchrocyclotron accelerators for proton hadron-therapy produce bursts of primary particles with a repetition rate up to 1 kHz and duration around 10 µs. Other technologies like laser-driven accelerators or free electron laser reduce the pulse duration down to fs with extremely low repetition rates. The need of accelerating primary particles in short and very intense pulses is even more pressing considering the emerging and promising FLASH radiotherapy [2]. The FLASH effect is induced by treating the cancer with an extremely high dose rate (above about 1 MGy/s) and allows obtaining the same effect on the tumour as with conventional radiotherapy, while damaging the surrounding healthy tissue to a lesser extent. The stray radiation field generated around these installations maintains the same time structure as the primary beam. Consequently, the commonly expected workplace fields are characterized by mixed and pulsed radiation components. The correct operation of active neutron detectors normally used in steady neutron fields, specifically rem-counters and BSS, can be hindered by pulsed fields because of high dead time losses. This effect limits the usability of active detectors based on neutron moderation up to few nSv/burst, which is quite low, especially when measuring inside the treatment rooms. This work describes the efforts of the EURADOS WG11 to define the problem, characterize instrumentation and propose solutions to face the issue of pulsed and mixed radiation fields. WG9 and WG11 coordinated their actions in intercomparisons for the characterization of mixed and pulsed fields. Despite the EURADOS initiative, open issues remain, especially concerning the metrological traceability, the optimization of available instruments and education and training.

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# Comparison between CLYC-6 and $^3$He for thermal neutron detection

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A $Cs_2LiYCl_6$ crystal, highly enriched in $^6$Li (>95%) (CLYC-6) was investigated for thermal neutron detection as an alternative to the commonly used $^3$He proportional counter, in view of the diminishing stockpile of $^3$He gas [1,2]. CLYC-6 was characterised in terms of thermal neutron detection efficiency and γ-ray rejection capability as compared with $^3$He.

All the measurements were performed with the thermal neutron source available at the CERN radiation calibration laboratory. The set-up consists of a cylindrical polyethylene (PE) moderator that surrounds an Am-Be source (888 GBq) and an optional PE reflector box enclosing the detector under test. The reference thermal neutron fluence at 30 cm distance from the source is $7.43 \times 10^2$ cm$^{-2}$s$^{-1}$ with the moderator only ("cylinder" configuration) and $5.75 \times 10^3$ cm$^{-2}$s$^{-1}$ when adding the reflector ("cylinder + reflector") [3]. The detectors under test were an 18 ×18 × 5 mm$^3$ CLYC-6 crystal grown by Radiation Monitoring Device and a spherical Centronic SP9/152/Kr $^3$He proportional counter (3.2 cm diameter, filled to 4 atm). CLYC-6 was coupled with a R6231-100 Hamamatsu Photomultiplier Tube and the output signal sent to a CAEN DT5720 digitizer (12 bit, 250 Ms/s). The $^3$He counter was supplied with +820 V and the output signal processed by an ORTEC 142IH preamplifier, an ORTEC 570 amplifier and an Amptek Pocket Multi-Channel Analyser 8000D.

Eight measurements were performed changing the source-to-detector distance from 30 cm to 70 cm with the two abovementioned source configurations. With CLYC-6 the thermal neutron counts were determined after a Pulse Shape Discrimination (PSD) analysis provided by the digitizer, and selecting the neutron signals detected under the thermal neutron peak, similar to [4]. With the $^3$He counter, the neutron counts were obtained integrating the acquired spectra above a threshold set to cut the γ-ray background. For both detectors the count rate decreased following the inverse square law. At each position, the efficiency in terms of count rate per unit detector volume was comparable.

Further measurements were performed placing each detector between the thermal neutron source and a 3 TBq γ-ray $^{137}$Cs source. Both detectors were first irradiated with the Am-Be source only and later simultaneously exposed to the neutron and the γ-ray sources. The aim was to evaluate their capability to detect thermal neutrons in the presence of a high intensity γ-ray field. At 30 cm distance from the Am-Be source (600 cm from the $^{137}$Cs source) the $^3$He response was not affected by the γ-ray field. With CLYC-6, the thermal neutron peak was identified from the PSD versus energy plot. However, the neutron count rate significantly reduced in the presence of the γ-ray field, because of the dead time induced by the high sensitivity of CLYC-6 to γ-rays. At larger distances from the Am-Be source (and closer the γ-ray source), the response of CLYC-6 for the "cylinder" configuration was dominated by γ-rays, while for the "cylinder + reflector" configuration the thermal neutrons were still detected but less efficiently. The $^3$He count rate was instead only partially affected by the high intensity γ-ray field, *i.e.* the count rate slightly differed (by only a few percent) in the presence of the $^{137}$Cs source.

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# Optimization of a new neutron detector based on liquid scintillator for neutron bursts related to thunderstorms


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The large difference of electrostatic charge in the thunderclouds can form a large particle accelerator which modifies intensity of secondary cosmic rays in the atmosphere and accelerates electrons and positrons up to the energies of tens of MeV. The bremsstrahlung generated by these high-energy electrons and positrons can often be measured on the ground and even by the satellites in space. The number of generated high-energy photons is large up to $10^{17}$. Bremsstrahlung photons can interact with air and soil by photonuclear reactions that generate neutrons. These phenomena have been mostly observed at high altitude observatories and places with very low altitude cloud bases such as winter storms in Japan.

The first observation of neutrons related to thunderstorm activity was reported already in 1985. Since then, many other measurements have been published. However, over the years it was shown that some previously used detectors had flaws that made them not suitable for such measurements. Probably the most important is the influence of electromagnetic interference that artificially enhanced the neutron count rates. Therefore, there are only very few measurements of thunderstorm neutrons which cannot be doubted.

We have designed a neutron detector capable of measuring fast and thermal neutrons generated by thunderstorm related phenomena. The detector is based on two liquid scintillators that support the pulse shaping discrimination and are also enriched with natural boron. The detector is intended to be placed at high-mountain observatory at Lomnický štít.

We will present the optimization of the neutron detector for such events mainly the optimization of pulse shape discrimination technique, calibration of photon and neutron components, in the field of the 14 MeV neutron generator at VSB-Technical University of Ostrava and algorithm for discrimination of $^{10}B(n,\alpha)Li^{7}$ reaction of thermal neutrons. The simulated response of the detector to such an event will be shown.

# Extended time-of-flight measurements down to 100 keV at the AMANDE facility with a stilbene scintillator

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ABSTRACT

In neutron metrology, the time of flight (ToF) is a primary method for the determination of the neutron energy, as it is directly related to length and time standards. The accelerator-based IRSN AMANDE facility at Cadarache (France) [1] is dedicated to neutron metrology and dosimetry in mono-energetic fields between a few keV and 20 MeV. AMANDE's ion beam can be pulsed [2], allowing time-of-flight (ToF) measurements to determine the neutron energy. Above 1 MeV, time-of-flight measurements are routinely performed with liquid scintillators such as the BC501A [3]. However, for energies below 1 MeV, the fluence references are generally only established with long counters [4]. Indeed, the efficiency of these liquid scintillators drops quickly and neutron/gamma discrimination becomes difficult. Even if most of the parasitic photons can be discriminated in ToF measurements at AMANDE, such discrimination is difficult in more complex mixed fields. Moreover, neutron fields generated by deuton ion beams of energy higher than the reaction thresholds (d,n) on carbon or oxygen [5] are of major interest and could be characterized with a dedicated low energy stilbene.

Recent developments of a new generation of stilbene crystals [6] have lead to detectors having both a good light output and a good Pulse Shape Discrimination (PSD). These detectors appear suitable for time-of-flight measurements, combined with neutron/gamma identification, at lower neutron energies than with liquid scintillators.

We have investigated an extension of the ToF method to low neutron energies using a 1"x1" solution-grown stilbene scintillator coupled to a digital acquisition system. Test measurements with this stilbene have been performed between 84 keV and 230 keV. The goals were to investigate the lowest reachable energy in the 80 keV to 230 keV energy range, as well as to determine the scintillator response time.
This study, supported by the French national metrological institute (LNE), is a part of a collaboration between IRSN and CEA [7]. The IRSN aims to find a detector able to cover the ISO-8529 [8] recommended neutron energy range between 144 keV and 19 MeV at AMANDE.

The ToF test measurements carried out at AMANDE [9] show that the stilbene remains efficient for energies down to 100 keV while having a good enough neutron / gamma discrimination as well as a fast response time. This work demonstrates the possibility to better characterize low-energy contributions in complex neutron fields

and thus improve calibrations for devices with a significant response between 100 keV and 1 MeV. However at least two detectors (or two settings) would be necessary to cover the whole energy range from 100 keV to 19 MeV.

Acknowledgment

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# Measurements of the secondary neutrons generated at the DRACO laser-driven proton accelerator

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Laser-plasma interaction can nowadays be exploited to accelerate very short and intense bunches of ions with energies up to dozens of MeV, within micrometer distances [1]. Mainly owing to the small spatial and temporal scale at which the acceleration process takes place, laser-driven ion sources show a great potential for becoming the next generation of particle accelerators, with various applications including hadron-therapy [2]. Due to the intrinsic large divergence of the laser-emitted particles, a fraction of accelerated ions is lost in the focusing and shaping apparatus, or on the vacuum chamber walls, resulting in the production of pulsed secondary radiation, mainly composed by neutrons, X- and γ-rays. A detailed experimental characterization of the secondary pulsed radiation expected in the proximity of laser-driven ion accelerators is essential when evaluating the feasibility of their future applications. The pulsed nature of the source, however, makes experimental investigations with most commercial active neutron detectors (such as conventional rem counters commonly used in radiation protection) troublesome, due to hardly correctable dead-time losses and pile-up phenomena in the acquisition electronics.

The Helmholtz-Zentrum Dresden-Rossendorf (HZDR) operates DRACO, a Ti:Sapphire PW-class laser system mainly used for investigations of relativistic laser-plasma physics, with applications in proton acceleration and beam optimization for radiobiological studies [3]. In August 2021, parasitic measurements of the produced secondary neutron radiation, performed by using various laser-target materials and laser pulse energies from 8 to 30 J (with pulse duration of ~30 fs), were conducted over a total of more than 300 laser pulses. For this, two neutron detectors were employed: the passive LINUS, a CR-39-based passive neutron rem counter (intrinsically unaffected by the pulsed nature of the radiation) and the LUPIN BF3-NP [4], an active neutron rem counter specifically conceived to work in pulsed radiation environments. Additionally, photon dose was also monitored by an ion chamber developed for pulsed photon fields. The neutron dose produced by each single proton bunch was detected by using the LUPIN BF3-NP. Differences in the production of secondary neutron and photon dose were found depending on the different laser-target materials employed. The total neutron dose of the whole measurement campaign was then benchmarked by the reading of the passive LINUS, showing reasonable agreement with the integrated measurement of the LUPIN BF3-NP.

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# Calculation and validation of the response function of a Bonner sphere spectrometer with a fission chamber as central thermal detector

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Neutron spectrometry measurements using Bonner sphere spectrometers are a key method for determining the neutron field in radiotherapy environments [1]. In order to achieve valid results, the central neutron sensor has to be chosen according to the measurement requirements. Neutron spectrometry in photon therapy environments needs to address the difficulty of performing measurements in mixed photon neutron fields. The Physikalisch-Technische Bundesanstalt (PTB) expanded the capabilities of its extended-range Bonner sphere spectrometer NEMUS [2] by incorporating a fission chamber as central thermal neutron sensor (CTNS) to adapt it to the measurement requirements of photon therapy environments.

An essential requirement for the operation of a Bonner sphere spectrometer is the detailed knowledge of the detector response functions. In this work, we extend previous approaches for response function calculations to provide a set of response functions for the extended-range Bonner sphere system NEMUS with a fission chamber as CTNS. To achieve this, an existing realistic model of the NEMUS Bonner sphere spectrometer was modified by adding a realistic model of the fission chamber as CTNS. The validity of this fission chamber model was established by comparing the calculated and measured thermal neutron responses based on measurements at the PTB thermal neutron field [3]. The Monte Carlo N-Particle (MCNP6) code was used to calculate the response of this Bonner sphere system to monoenergetic neutrons in the energy range 1 meV to 100 MeV.

To demonstrate the validity of the set of response functions for the Bonner spheres, we analyzed a set of validation measurements performed at the monoenergetic neutron facility PIAF at PTB. Calibration measurements with monoenergetic neutron beams with neutron energies of 0.565 MeV, 1.25 MeV, 2.5 MeV and 14.8 MeV for selected Bonner spheres were used to compare the calculated response functions to the measured values.

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## Development and validation of a model for assessing neutron fluence with lithium fluoride thermoluminescent detectors

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Lithium fluoride thermoluminescent detectors are commonly used for neutron measurements. In this case, the detector calibration is usually performed with readily available gamma sources in terms of gamma equivalent air kerma instead of the more meaningful neutron fluence. However, the neutron calibration coefficients in terms of neutron fluence depend strongly on the experimental conditions. When performing measurements with a certain type of thermoluminescent detector in a certain neutron field, usually the calibration coefficient applicable for that specific case cannot be found in literature. Therefore, the goal of this work was to develop and validate a model to calculate such neutron calibration coefficients and to use this model to quantify the main sources of uncertainty.

A model was developed to predict neutron calibration coefficients based on analytical calculations and Monte Carlo radiation transport simulations in MCNP6.2 and PHITS. This model takes into account the thermoluminescent detector type (dopants, thickness, density and $^{6}$Li fraction), the neutron energy and angular distribution and the luminescence reader light collection geometry. Neutron calibration coefficients were also assessed experimentally by irradiating different lithium fluoride thermoluminescent detectors at the Belgian Reactor 1 at SCK CEN with the Z55B thermal neutron beam.

The performance of the model was evaluated by comparison of the neutron calibration coefficients predicted by the model with experimental data from literature and this work. This comparison showed reasonable agreement. However, a slight systematic deviation was found, probably caused by the significant uncertainty on the relative luminescence efficiency values predicted by the recently developed Microdosimetric d(z) Model for the low energy charged particles released during the $^{6}$Li(n,$\alpha$)$^{3}$H reaction. Average relative luminescence efficiency values based on the experimental results were also derived and compared with the theoretical results.

The neutron energy, neutron angular distribution and the geometry of the luminescence reader light collection were identified as the main sources of uncertainty. Imprecise knowledge of these experimental input parameters can easily lead to uncertainties of the order of tens of percent on the calculated neutron calibration coefficients. The developed model is applicable for neutron energies below 0.01 MeV, but could be extended to higher energies in the future.

# Assessment of a new extended passive neutron monitor based on TLDs with application in proton therapy centers and research facilities

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In medical and research facilities endowed with accelerators where stray neutron fields are large enough, as those yielded in proton therapy centers, it is necessary to verify operational magnitudes, like ambient and personal equivalent dose among others, with the purpose of assure the operational radiation protection of workers and public. Although these facilities usually have active devices to measure radiation from neutrons, it is highly advisable, and safe, to support this information with reliable and complementary measurements, being passive monitoring a key and critical task to assess the neutron exposure. Evenly, these experimental data from passive monitors could help to verify and confirm some assumptions used in the calculations of shielding and activation studies, always in the conservative side. Furthermore, neutron devices used in medical proton centers and research facilities with accelerators should be able of measuring both, high-energy neutrons and pulsed neutron fields. The aim of this work has been to carry out experimental measurements to verify the response of a new extended passive neutron area monitor, which use pairs of thermoluminescent dosimeters, and it is suitable for neutrons sources with energy range from thermal to hundreds of MeV. The device has been developed by Universidad Politécnica de Madrid, and its theoretical response was previously validated, through Monte Carlo codes, considering different irradiation geometries and configurations, so that the monitor can work with or without a spallation material layer, depending on the neutron field and radiation features at each place. The passive monitor has been tested in several facilities with accelerators yielding neutron fields, comparing results with both, theoretical response obtained with codes, and experimental data reached with others conventional active rem-meters.

**Keywords:** *Neutron area monitoring, Extended passive monitors, Proton therapy*

# Analysis of regional retention of plutonium in the respiratory tract of four acutely-exposed workers using scar-tissue compartments

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Animal studies and autopsies of former plutonium (Pu) workers have demonstrated that a small fraction of inhaled Pu may be retained in the respiratory tract compartments long after an inhalation intake, even when the inhaled material is known to be very soluble. The Human Respiratory Tract Model (HRTM) described in Publication 130 [1] of the International Commission on Radiological Protection (ICRP) provides some mechanisms to account for retention of material that can be subject to little to no mechanical transport or absorption into the blood. One of these mechanisms is deposition of the material in the sequestered compartments, and the other is 'binding'. The latter refers to a process by which a fraction of dissolved material chemically binds to the tissue of the airway wall. This fraction ($f_b$) is taken to be 0.2% for Pu [2], but given the implications of chemical binding for doses to the target tissues in the respiratory tract, it is important that this parameter be further evaluated and studied.

Data on regional retention of Pu in the respiratory tract of four workers – who had inhaled materials with solubility ranging from soluble nitrate to very insoluble high-fired oxides – were obtained at the United States Uranium and Transuranium Registries. Significantly more Pu was found to be retained in the upper respiratory tract of these workers than was predicted by the current biokinetic models [1, 2]. Modification of the model parameters, including $f_b$, was unable to explain the data for one individual who inhaled an insoluble form of Pu. Moreover, several studies show evidence of retention of a large amount of Pu in scar tissues of humans and experimental animals, pointing to an alternate mechanism by which Pu can be retained infinitely in the respiratory tract.

This presentation proposes a HRTM that is modified with the addition of scar-tissue compartments to describe the long-term regional retention of Pu in the respiratory tract of all four individuals. The transfer rates to the scar-tissue compartments were determined using Markov Chain Monte Carlo analysis of the bioassay and post-mortem regional retention data, taking into account the uncertainties associated with deposition, dissolution, and particle clearance parameters. The estimates obtained from modelling these data showed that a significant amount – 20-100%, depending on the solubility of the material inhaled – of Pu retained in the respiratory tract was sequestered in the scar tissues. Unlike chemically-bound Pu that irradiates sensitive epithelial cells, Pu in scar tissues may not be dosimetrically significant because the scar tissues absorb most, if not all, of the energy from alpha emissions.

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# Liposomal formulations of new decorporation molecules for the treatment of internal Strontium/Cobalt contaminations

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Nuclear/radiological incidents or accidents involving internal contaminations with radioactive cobalt or strontium compounds occur regularly. In this case, the current recommended therapeutic options (calcium salts of DTPA and cobalt gluconate in case of internal cobalt contamination and ammonium chloride and calcium gluconate in case of internal strontium contamination) are of low specificity, and their effectiveness remains modest. There is therefore a crucial need to design new modalities of administration and/or new molecules based on drug discovery or already marketed drugs for decorporation.

Our team is developing new liposomal formulations to deliver strontium/cobalt potential chelating agents to the main radionuclide retention organs. In a first step, we have identified and selected bisphosphonates (BP) molecules that can form stable complexes with cobalt and strontium. For that purpose, we have developed different approaches such as UV/Visible spectrophotometry and ion chromatography coupled with ICP-MS detection to screen our preselected candidates. These techniques have enabled the effective visualization of the BPs and divalent cations but compelling detections of BP-metal complexes have yet to be achieved. Additionally, we are currently developing a new separative analytical method based on capillary electrophoresis that should also be very useful for characterizing complexes.

The further step will be to encapsulate the candidate molecule into liposomes that will be surface functionalized with specific targeting ligands. The molecular targeting should induce a preferential accumulation of the liposomes in the bone compartments as well as in the liver and kidneys, thus enhances the distribution of the BPs in these key radionuclide retention organs. In vivo evaluation of the decorporation efficiencies of both the free and liposome-encapsulated molecule will then be performed in a rodent model.

## Neutrons Emitted by Alpha-Particle Reactions and Internal Dosimetry: Does It Matter?

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In the International Commission on Radiological Protection (ICRP) Publication 133, specific absorbed fractions (SAFs) for spontaneous fission (SF) neutrons are provided for 28 radionuclides. SAFs are defined for radiations emitted directly in the decay of radionuclides and include any absorbed energy subsequently deposited by any secondary particles created. So SAFs for SF neutrons are identified specifically as SAFs. Neutrons can also be produced by the reactions of alpha particles emitted by the radionuclide as they range out in organs and tissues. The question has arisen as to whether these neutrons are a missing source of equivalent dose which should be considered in obtaining target organ doses. In this work the authors have computed ($\alpha$,n) yield in elemental compositions corresponding to all source organs in the body for Pu-238, Ra-226 and Cf-252 decay. The yield of neutrons per Bq and the energy distribution within each source organ were generated using the SOURCES-4C code. As expected the ($\alpha$,n) neutron yield increases with alpha particle energy, and the nuclides which are principally responsible for the neutron production due to this reaction are C-13, O-17, O-18, Cl-27 and Na-23. As an example, the average energy of the neutrons from this reaction in marrow ranges from 2.41 MeV for the clavicle to 3.89 for the medullary cavities, corresponding to increasing carbon-to-oxygen ratios from 0.133 to 2.66, respectively. The resulting contributions to target dose will be presented for these radionuclides and, for the radionuclides that also spontaneously fission, compared with the equivalent dose due to SF neutrons on a per neutron emitted basis. To conform to the current ICRP formalism, the equivalent dose due to these neutrons must be embedded in the alpha particle SAF. Those values will be presented and discussed.

# Improving the reliability in the internal dosimetry of uranium workers. Application of the OIR uranium model to long-term occupational intakes.

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The International Commission of Radiological Protection (ICRP) has published in the OIR (Occupational Intakes of Radionuclides) report series an update of the biokinetic models and dosimetric data associated to internal exposures of workers consistent with ICRP Publication 103.

This work focuses on the application of OIR uranium model published in ICRP Publication 137. A better knowledge of the industrial uranium compounds allows a better characterization of their absorption in the body from the respiratory tract after inhalation, classifying OIR materials as Types F, M, S and intermediate F/M and M/S. The new OIR doses for uranium workers mean a reduction from the former ICRP 60/78/119 approach, except for Type S (very insoluble) compounds. The most relevant impact concerns exposures to uranium oxides, assumed formerly Type S when inhaled, but in the ICRP Publication 137 $UO_2$ and $U_3O_8$ become Type M/S material, which means more solubility, higher urinary excretion rates and lower dose coefficients.

New OIR dose coefficients for uranium mixtures are calculated for different absorption types, e.g. resulting in $e(50)_{OIR}$= 5.4E-6 $SvBq^{-1}$ for Low Enriched Uranium (LEU, M/S).

OIR uranium retention/excretion models and dose coefficients were implemented in BIOKMOD code (http://oed.usal.es/webMathematica/Biokmod), a dosimetric tool which allows the interpretation of monitoring data in terms of intake (Bq) and Doses E(50) Sv. Retention and excretion values in BIOKMOD and dose coefficients were validated comparing with these from ICRP Data Viewer (Publication 141).

CIEMAT and University of Salamanca in Spain have been working on the re-interpretation of the in vtro bioassay data from 100 workers long term exposed to inhalation of uranium oxides during the fabrication of nuclear fuel elements using LEU. The individual monitoring program of this workforce consists of measurement of uranium in 24h urine samples by alpha spectrometry on annual or semi-annual basis for a chronic intake scenario. Workplace monitoring using SAS (Static Air Sampling) complements individual dosimetry.

E(50) doses were re-assessed by applying the new OIR excretion model and dose coefficients for Type M/S uranium materials, considering acute and/or chronic intakes resulting in a reduction by roughly a factor of 4 comparing with former doses based on ICRP 60/78/119.

The improvement of the reliability of the uranium doses here presented is also based on the study of the natural background due to the diet in 70 workers never exposed to uranium oxides who provided urine samples before starting exposure in the nuclear manufacturing facility. Background levels of 0.35 $mBqd^{-1}$ of $^{238}U$ and 0.57 $mBqd^{-1}$ of $^{234}U$ in urine were found, showing a $^{234}U/^{238}U$ ratio of 1.6, in agreement with Hurtgen et al, 2001.

The use of the SAS data for reducing the uncertainty associated to the time and random nature of chronic uranium intakes at the workplace also means an important improvement on dose evaluations.

# Contact restriction time for nuclear medicine patients: a practical method and calculation tool

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Nuclear medicine patients are a source of exposure for members of the public, their relatives and co-workers. In Europe and in the US intsructions have to be delivered to those patients before their release in order to limit the public exposure. These instructions prescribe to avoid contact with categories of people, like co-workers, children, pregnant women, for a given period. These instructions are necessarily based on exposure scenarios that define the contact time and distance for each category of contact person.

Several methods have been proposed to deliver these instructions and general guidelines have been issued. The general guidelines are usually based on the activity administered to the patient and thus ignore the high variability of whole-body retention and thus of the dose rate. The methods published up to now are formally correct but do not come with easy to use associated tools. Moreover, they usually consider crudely the dose rate variation with distance.

A general method that calculates contact restriction time has been implemented in a spreadsheet. The spreadsheet first defines well accepted contact scenario parameters (time, distance, dose limit) but the user can set-up its own.
Then, the calculation is performed according to the following hypotheses:

- the whole-body retention is described by a bi-exponential function.
- the dose rate is proportional to the whole-body retention, normalized by a dose-rate measurement.
- the dose rate varies with distance according to the patient line source model.

The restriction time for a contact scenario is the shortest date so that integration of the dose rate from that date to infinity is below the dose limit.

The first hypothesis implies that for a given therapy the retention function is known. From the literature we recommend default values for $^{131}$I treatment (benign disease and remnant ablation) and for $^{177}$Lu therapy.

The second hypothesis requires that a dose rate measurement is performed at the patient release date and entered by the user in the spreadsheet. This measurement can be performed at any distance, but one meter is a reasonable choice.

The third hypothesis has been tested against published literature data as extensively as possible. Assuming a default height of 176 cm for the patient induces acceptable uncertainties for the dose rate variation with distances.

The method has been tested against published values of restriction time calculation. Results do not differ significantly and can be explained by the difference in calculation method or underlying hypotheses.

# Environmental Airborne Tritium Monitoring System based on Absorption of Tritiated Water on Calcium Chloride

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Environmental monitoring of radioactivity around nuclear facilities is conducted in order to demonstrate compliance with regulatory requirements on the protection of members of the public. Tritium release to the environment is considered to be common and characteristic of many nuclear facilities and therefore multiple systems and techniques are used to monitor its concentration in air, soil and flora.

A new monitoring system was developed at the Nuclear Research Center Negev (NRCN), which is based on the absorption of tritiated water on calcium chloride grains and analyzing its absorbed activity using a liquid scintillation system. Calcium chloride has an improved capability to absorb water vapor, as compared to other absorbents/adsorbents that are common in tritium monitoring: it can absorb water vapor as much as its weight.

After the determination of the tritium activity absorbed in the calcium chloride, a model is used to estimate its airborne concentration, considering the environmental conditions such as temperature and relative humidity.

# Radioiodine in thyroid calibration for the in vivo measurement of exposed population in emergencies

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In emergencies, due to an accident in a nuclear medicine service or in a nuclear power plant, 131I can be released with the consequent risk of internal contamination of workers and population. In such scenarios, it is necessary to develop fast and effective methods in order to assess the internal exposure for exposed persons guaranteeing the reliability in dose assessment especially for children.
CIEMAT Whole Body Counter has implemented new calibrations to determine 131I in the thyroid for different age groups (1, 5, 10 and 15 year-old children and adults).
A set of neck phantoms for children was designed and manufactured by CIEMAT; thyroid glands were simulated using vials filled with a liquid solution of 131I, based on ICRP Publication 89.
One LE Ge and other BE Ge detector inside a shielded room, and a Fastscan counter with 2 NaI(Tl) detectors were previously calibrated for in vivo measurement of radioiodine in thyroid for an adult male. Analyzing the measurement of children with the efficiency of an adult male, there is a bias of 35% of the activity for the youngest child with the LE Ge detector, and 65% in the Fastscan counter when comparing with activities obtained using age-specific efficiencies.
Counting geometry of the new thyroid calibrations for LE Ge detector and BE Ge one consist of a neck phantom centered below detector, at a distance of 15 cm. In case of Fastscan, phantoms are positioned parallel to NaI(Tl) detector at a distance of 12 cm. Efficiency curves vary with energy and depend on phantom and source sizes.
Detection limits of 131I are in the range of 26 - 42 Bq for Fastscan (counting time of 300 s), in the range of 4.6-6.2 Bq for LE Ge detector (1200 s), and between 6 to 12 Bq for BE Ge detector (600 s).
Different calibrations for adults and children allow measuring realistic activities and improve an early emergency response. Fastscan counter is useful in case of a single contamination with 131I, and Germanium detector is better to be used in case of complex contamination due to its excellent resolution.

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# Individualized dosimetry with planar imaging in 177Lu therapy

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Although fully individualized dosimetry is at least advisable in radionuclide therapy, to date personalized dosimetric evaluation is not a routine practice in most of the hospitals, probably because detailed dose estimation in nuclear medicine therapies is highly demanding in terms of machine and personnel time, and generally implies longer hospitalization and repeated visits for patients.
We have developed [1] an approximated calculation method for the evaluation of doses received by radiosensitive, non-targeted organs and tissues, based on an automatic segmentation algorithm and the computational phantom developed within the OpenDose project. Originally conceived for NEuroendocrine Tumours (NET) therapies with somatostatin-analogues radiolabelled molecules, our approach has been validated in different frameworks, demonstrating its applicability into therapies with different radiopharmaceuticals.
Starting from a single Whole-Body (WB) scintigraphic image, and working similarly to more accurate methods based on repeated WB scans [2,3] our approach has given interesting results in bone marrow dosimetry, although with a degree of approximation with respect to calculations including at least one SPECT scan.
The method and its capabilities of correctly describing doses imparted to different anatomical districts will be compared with classical individualized dosimetric calculation schemes. Perspectives on the future developments of the approach and its applicability to large cohort of patients, involved in therapies with different radionuclides, will be discussed.

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# The MARE experiment – phantom radiation measurements during the NASA Artemis 1 mission to the Moon and back

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Cosmic radiation constitutes one of the most important hazards for humans in long-duration space missions. The assessment of the radiation risk to astronauts requires knowledge of the radiation environment and doses accumulated in the body. For flights reaching beyond the low Earth orbit (LEO), e.g. to the Moon orbit, such knowledge is very limited and insufficient. In the near future, a space mission is planned, which will help to fill partly the gaps in our knowledge.
The NASA Artemis-1 mission of the Orion spacecraft is planned to be launched in early 2022. Orion will cross the Earth´s Van Allen radiation belts and spend some time in an orbit around the Moon. The total mission duration will be between 25 and 42 days. The Orion vehicle is designed to carry four astronauts, but the Artemis-1 flight will be unmanned. This situation created an absolutely unique opportunity for conducting a large-scale dosimetric experiment during this first flight. Within the MARE experiment (Matroshka AstroRad Radiation Experiment [1], organized by NASA, ISA, StemRAD, Lockheed and DLR) two anthropomorphic female phantoms named Helga and Zohar will be placed on their respective crew seat inside Orion. Zohar will be shielded with a specially designed protective vest (AstroRad, developed by StemRad, Israel), while Helga will be unprotected. The phantoms will be equipped with a large number of active and passive radiation detectors. These detectors will be distributed throughout the phantom volumes, as well as on the phantom surface, which will allow reconstructing a 3D distribution of radiation doses and other important parameters of the radiation field.
The MARE experiment is conducted under the scientific leadership of the DLR Institute of Aerospace Medicine in Cologne (Germany) and with the participation of research teams from Europe, Japan, and the USA. MARE is built on the heritage of the Matroshka experiment [2,3] and the experience gathered from the ongoing DOSIS 3D project on the International Space Station [4]. The presentation will provide an overview of the MARE experiment and the used detector systems.

**Acknowledgments:**
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# Modelling the neutron biological effectiveness on the Mars surface

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Radiation represents a major issue for human space exploration and big efforts are being dedicated to assess the risk to health and determine countermeasures to minimize the damage on astronauts. Among all radiation qualities, neutrons play a critical role: though neutron physical dose to astronauts might be small, the biological effectiveness of neutrons varies a lot with their energy and can be relatively high. In the space radiation environment, neutrons are not present as a primary radiation, but they are largely produced in a wide energy range by the interaction of the primary radiation components with, e.g., the planetary atmosphere or soil, space vehicle walls or body tissues. Modelling is of fundamental importance to characterize the biological effectiveness of neutrons in space, since it is almost impossible to reproduce the neutron component of space radiation in ground-based experiments.
In this framework, we further refined and developed a model of Relative Biological Effectiveness (RBE) of neutrons for the induction of DNA damage as a function of neutron energy in the range $10^{-6}$–$10^{5}$ MeV [1]. We used the transport code PHITS to simulate the exposure of an ICRU44 soft tissue spherical phantom, representative of the human trunk, immersed in an isotropic field of monoenergetic neutrons, and new analytical functions reproducing DNA damage obtained with the biophysical code PARTRAC [2,3] for secondary charged species. The spherical phantom, with radius of 15 cm, has been divided into 15 isocentric shell regions. The RBE evaluation in these regions, thus as a function of depth, makes it possible to link a specific value of the biological effectiveness of neutrons with a given incident energy to a specific organ/tissue. As an application to neutrons in space, we finally aimed at a quantitative evaluation of neutron-induced biological effects to an astronaut on the Mars surface. To this purpose, RBE values need to be associated to the dose absorbed by the astronaut in this scenario. Such dose has been determined by simulating the exposure of the phantom to neutron energy spectra due to interactions of galactic cosmic rays with Mars atmosphere and soil, obtained with the GEANT4-based code AtRIS. The convolution of the physical dose with the RBE at different depths gives the so-called RBE-weighted dose, which provides a good indicator of the neutron biological effectiveness in the Mars exposure scenario.

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# A new compact spectrometer for neutrons produced by cosmic rays


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The development of modern compact neutron spectrometers remains of topical importance for use in a wide range of contexts outside the laboratory, including personal and workplace dosimetry. In particular, for neutron dosimetry in the upper atmosphere and in space, measurements in the energy range 1–120 MeV are necessary since there is a significant peak around 100 MeV in the energy spectrum of secondary neutrons produced by cosmic rays [1].

We report of progress on the development of new compact neutron spectrometers [2] based on EJ-276 plastic scintillator, SensL C-series silicon photomultipliers (SiPM), and digital pulse processing, including the implementation of pulse shape discrimination to separate detector events associated with neutrons and gamma rays. Neutron energy spectra are produced from measured light output spectra using unfolding analyses based on known response functions for the device, either from measurement or simulation. The first version of the compact spectrometer consists of a scintillator of dimension 6×6×50 $mm^3$ coupled to a single SiPM, which is suitable for measurements of neutron energy spectra, and absorbed dose, up to 20 MeV using response functions produced using Geant4. A second version, suitable for measurements up to 120 MeV, makes use of a larger scintillator coupled to two SiPMs. With increasing neutron energy, the proportion of recoiling charged particles which escape from the scintillator volume also increases, distorting the response functions of the spectrometer. Furthermore, neutron interactions with carbon nuclei in the scintillator become more important at higher energies, with many reaction channels possible. The present deficiencies in the cross-sectional data for n-C interactions result in unreliable detector response functions derived solely from calculation, hence measurements are necessary. The fast neutron facility at iThemba LABS is ideal for such purposes since ns-pulsed neutron beams are available with the quality necessary to characterise bespoke detector systems of this type. We report on direct measurements of response functions and subsequent unfolding analyses across the full energy range of interest and discuss deployment of the device for in-flight testing.

The project is associated with the newly established South African Space Neutron Initiative (SASNI).


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# A Bonner Sphere spectrometer for the SAMADHA project

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**Abstract text:**

The SAMADHA project (South Atlantic Magnetic Anomaly Dosimetry at High Altitude) of INFN will study the South Atlantic Magnetic Anomaly by setting up measurement stations at high-elevation laboratories in different places in South America, such as Chacaltaya in Bolivia (5240 m a.s.l.).

As far the neutron dosimetry is concerned, an extended range Bonner Sphere spectrometer is under construction for this purpose. As optimal trade-off between cost and sensitivity, cylindrical Helium-3 detectors (10 bar x 2.8 $cm^3$) will be adopted as central thermal neutron detectors.

This work describes Bonner Sphere spectrometer for the SAMADHA project with focus on the central detectors, the response matrix and the preliminary measurements performed for its validation.

# Angular calibration of PH32 silicon strip detectors at HIMAC

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Strip detectors are planned to be utilized in future experiments for impact angle estimation of interacting particles from cosmic radiation or during thunderstorms onboard aircraft or UAVs and also as a part of the more advanced hybrid LET spectrometer for measurements of cosmic radiation onboard spacecraft. For all these applications, it is necessary to precisely measure angles of incoming particles. In this work, it is shown that silicon strip detectors developed by CTU in Prague and esc Aerospace company, fulfil specific requirements for angle measurements. Optimal setup of parameters for processing the data given by strip detectors is described and selected.

The strip detector consists of four silicon strip sensor layers read-out by PH32 ASICs. There are two PH32 chips side by side per one layer, which gives a detection area almost 2x2 cm$^2$. On each layer, there are 64 electrodes in strip form on the surface of two 525 µm thick silicon sensors. Each layer is rotated 90° relative to the previous layer. This arrangement provides coordinates of spatial points, making possible the estimation of the angle of incoming particle (if the particle passes through all four layers of the detection stack).

The strip detector was irradiated by several ions (H, He, C) from various directions. The measurement was performed at the Heavy Ion Medical Accelerator in Chiba (HIMAC). Based on data from these measurements, we demonstrate a new method for calculating the angles of detected particles. Results will be compared with angle values expected from the geometry of performed experiments. General properties, advantages, disadvantages and limitations of examined silicon strip detectors will be discussed.

# Stochastic Properties of Radiation Tracks in LiF Fluorescence Nuclear Track Detectors

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Lithium fluoride (LiF) is a well-known luminescent and optical material recently applied to detect and visualize tracks of ionizing particles [1]. It became possible to measure at IFJ PAN under fluorescent microscopy the photoluminescence emission from a single densely ionizing particle in LiF crystals with the resolution of about 0.5 μm. Such tracks were observed following exposure of LiF crystals to ion beams, $\alpha$-particles, neutrons and even electrons and γ-rays [1]. The tracks created by fully ionized high-energy nuclei form straight lines reproducing the actual path of particles in matter. These lines with local minima and maxima are not continuous but broken by places without measurable fluorescent emission ("gaps"). The effect is stronger for low-Z ions characterized by lower stopping power. The physics of this effect is not clear since the average energy deposited by ions is high and, at the resolution of about 0.5 μm, should lead to an uniform fluorescence emission along the track. The gaps are also visible in other track detectors, including FNTD detectors based $Al_2O_3$:Mg.

In this research the fluctuations of energy deposition along the particle tracks were analyzed in order to explain the observed effect of gaps. The optical density distribution along heavy charged particle tracks were measured at LiF crystals. The distributions are partly following energy loss distributions as calculated from of Vavilov formula [2]). Monte Carlo track structure codes were next applied to calculate the stochastic of energy deposition in heavy ion tracks at volume size ranging from a few nanometers to a few micrometers. Monte Carlo particle transport codes were applied to simulate individual ionizations and excitation in the track structure of the heavy ions. The simulations were performed with nanometer resolution in wide range of volume size (from few nanometers up to tens of micrometers)to reveal the stochastic nature of energy depositions patterns. The effect could be partly explained by fluctuations produced by delta-rays traveling outside the direct vicinity of the track.

**Acknowledgments:**
This work was partly supported by the National Science Centre, Poland (grant No 2020/39/B/ST9/00459).

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# Calibration of a gamma camera for RN emergency preparedness

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The Fukushima Daiichi accident in Japan 2011 showed a need for extensive whole-body measurements for estimation of internal contamination following a radiological and nuclear emergency [1]. Although whole-body counters (WBC) are commonly used for this purpose, the access to WBC in Sweden is limited (around 15 units) and other resources should be investigated.
Medical gamma cameras have shown to be feasible for estimation of internal contamination as an alternative to WBC [2,3,4,5] Gamma cameras have good geographical spread, are comparatively abundant (about 75 in Sweden) and used on a daily basis in the clinic, which means that trained personnel are available for the measurements. The aim of this study was therefore to identify the most common gamma camera models in Sweden and to develop a method for using one of the most common models in the Swedish RN emergency preparedness with a minimum of training and adjustment of the clinical equipment.

A gamma camera (General Electric's Discovery 670 NM/CT PRO) at the Sahlgrenska University hospital in Gothenburg, Sweden, was calibrated for $^{152}$Eu, $^{137}$Cs, $^{60}$Co, and $^{40}$K. This model was chosen due to its abundance, based on a survey of gamma camera models used in nuclear medicine departments at Swedish hospitals. It consists of dual head detectors (NaI) with a thickness of 5/8". The calibration was done for lying and sitting geometry using the IRINA whole body phantom [6], covering body sizes of 12 to 110 kg (lying geometry) and 12 to 50 kg (sitting geometry). All measurements were made without collimator, to enhance the detection efficiency [2,3], using the total number of background corrected counts in the energy window.
The minimum detectable activity (MDA) showed that this gamma camera model can be used to measure activities resulting in an effective dose well below 1 mSv for all phantom sizes, for $^{152}$Eu, $^{137}$Cs and $^{60}$Co [7,8,9].

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# *In vivo* public monitoring in emergency exposure scenarios by means of spectrometric and non-spectrometric devices

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In a nuclear or radiological accident scenario, when potentially members of the public can undergo internal contamination by the anthropogenic radionuclides released in atmosphere, effective methods that can be used directly in the field to perform fast scan for internal contamination on a large number of individuals can play a major role to undertake appropriate countermeasures. Specific attention should be paid on the individual monitoring of children, since they constitute the sensitive population group with the highest risk of developing cancer. At the ENEA Casaccia Research Center in Rome (Italy) monitoring procedures based on a portable HpGe detectors and ratemeters have been tested in the field. Such devices have been calibrated using both a BOMAB (BOttle Mannikin ABsorption) phantom, spiked with a known amount of a mix-radionuclide liquid source, and neck phantoms equipped with pairs of vials of different sizes to simulate the thyroid lobes for different ages, spiked with $^{131}$I and $^{133}$Ba. The detectors have been tested to evaluate the procedure sensitivity performing measurements with the WBC spectrometric equipment for less than 200 seconds and with the non-spectrometric equipment for thyroid scans in 60 seconds. A total of 170 acquisitions of uncontaminated volunteers for the blank measurement have been collected, in particular 105 acquisitions on adult and 65 acquisitions on 10 y/o children. The detection limit calculation, in terms of activity taken up (Bq), has been carried out according to the Standard ISO's 11929 and 28218 whereas the corresponding "minimum effective dose" (mSv) have been calculated on the basis of ICRP Publications 134, 137 and 141. The measurement campaign has been performed in open field with a high background (ambient dose equivalent H*(10) rate more than 0.2 μSv/h), in order to study both the technique potentials and limitations. Results and evaluations of the measurement campaign are presented and discussed with a particular focus on the peculiar features of this technique in respect to the aspects reported in current literature.

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# INITIAL INDIVIDUAL DOSE ASSESSMMENT FOLLOWING A NUCLEAR ACCIDENT

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Introduction
In the early phase of a nuclear reactor accident, *in-vivo* monitoring of impacted population would be highly useful to detect potential intake during the passage of the cloud and to estimate the dose from inhalation of measured radionuclides. However, it would be important to take into account other exposure components: (1) inhalation of unmeasured radionuclides and (2) external irradiation from the plume.

Methods and results
Here is presented a methodology to calculate coefficients used to convert *in-vivo* measurement results directly into doses, not only from the measured radionuclides but from all sources of exposure, by combining accidental model-based projected doses with the results of the *in-vivo* measurements of a tracer radionuclide.
As an illustration, the methodology is applied to two scenarios of accidents affecting a nuclear power plant (Pressurized Water Reactor type): a loss-of-coolant accident (LOCA) leading to core meltdown and a steam generator tube rupture (SGTR) accident. Calculated coefficients are directly applicable to perform a first dosimetric assessment of the exposure for a person with $^{131}$I measured in the thyroid in the aftermath of an accident related to one of the two scenarios considered (i.e. with similar activity releases and dispersion conditions).

Conclusions
Early interpretation of *in-vivo* measurement, as presented, gives crucial indication on the real level of exposure of the population around the nuclear site. It allows to communicate with the measured persons on the basis of individual measurement results. A more sophisticated dose assessment, taking into account the local isotopic composition of the contaminated cloud, the time-pattern of the exposure, subsequent measurements if any and possibly the post-accidental exposure, will have to be performed in a second time in order to evaluate individual health risk for long term medical survey and to provide a quantification of the exposure for epidemiological studies.

# Reference radon chamber at Politecnico di Milano: characteristics and future perspectives

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The publication of the Council Directive 2013/59/EURATOM has highlighted the problems related to radon exposure and it establishes reference levels for indoor radon concentrations. The directive was included in Italian legislation in 2020, and asks for stringent requirements on laboratories that carry out measurements of radon in air. Among these, there is the request of periodical instrument calibration and participation to inter-comparison tests. In this context, the need of reference calibration facilities (STAR or radon chambers [1]) is becoming pressing, both for metrological traceability and metrological confirmation.
At the Radiation Metrology laboratory of the Politecnico di Milano, a 2 $m^3$ radon chamber has been operating since 2016 for instrument calibration and radon device exposures. The accreditation process began in 2019, and it was completed in March 2021.
In this work, the authors present the characteristics of the radon chamber, the measurements for its characterization (radon leak rate, spatial uniformity of the concentration and gamma dose rate), the metrological traceability, including uncertainty assessment and the procedures to ensure the long-term stability.
A further focus is on using radon chamber for inter-laboratory comparison exercises. In early 2021, a proficiency testing for CR-39 based devices was organized and specifically conceived to evaluate the capability of laboratories to manage the effects of aging and fading; the test end is fixed in April 2022.

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[1] IEC 61577-4. Radiation protection instrumentation - Radon and radon decay product measuring instruments - Part 4: Equipment for the production of reference atmospheres containing radon isotopes and their decay products (STAR). (2009)

# In vivo monitoring at CIEMAT Whole Body Counter of a person contaminated with $^{177}Lu$ through a wound

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In case of radiological emergency or incidents at the workplace, internal contamination through wounds may occur with the resulting passage of the radioactive material into the bloodstream. The dose assessment may be based on the measurement of the activity of x-ray and gamma emitting radionuclides deposited in the wound entry taking into account the contaminated surface, wound depth and solubility of the chemical compound.

The aim of this work is to present a new procedure developed by the CIEMAT Whole Body Counter (WBC) for in vivo measurement of radioactive contamination through a wound using a LE Ge detector.

CIEMAT WBC has participated in an intercalibration/intercomparison exercise organized by IRSN with the aim to standardize the practice for the measurement of wound contamination in a puncture geometry. The wound calibration phantom was designed simulating internal contamination in human skin and was manufactured using tissue-equivalent PMMA circular radioactive sources (100 mm x 2 mm) of $^{241}Am$, $^{57}Co$, $^{137}Cs$, $^{133}Ba$, $^{85}Sr$ and tissue-equivalent PMMA blank layers simulating contamination on the surface (0 mm) or inside the wound (10, 18 mm).

The CIEMAT efficiency curves depending on the depth of the wound contamination showed that at energies below 250 keV, efficiency varies substantially at different depths in the wound while for energies beyond 250 keV efficiency variation is less meaningful. The detection sensitivity obtained allows to measure activities of a few Bequerels of the radionuclides of interest at short times of measurement.

This methodology was validated using a different wound source for the identification and quantification of the radionuclides present in this new IRSN wound phantom. The results confirmed proper bias when comparing with the reference activities of identified radionuclides ($^{241}Am$, $^{57}Co$, $^{137}Cs$).

The measurement of a CIEMAT radioactive point source simulating puncture contamination in wound (0 mm depth) was carried out using the efficiency calibration mentioned above, obtaining an accurate activity value of the source, demonstrating the feasibility of an alternative method of calibration using radioactive point sources simulating a surface contamination at the wound site (0 mm depth).

Measurements of a worker contaminated with $^{177}Lu$ through a puncture wound were performed at CIEMAT WBC using 2 BE Ge detectors. Direct measurements on the left hand's middle finger over the wound site were carried out considering a point source geometry and a detector-finger distance of 10 cm. Additional in vivo measurements in total body geometry were performed using a NaI(Tl) detector, shielding the wound area for a better estimation of the internal dose. The activity in the wound site was 2.7E+05 Bq; the activity in the total body was 5.6E+05 Bq, both obtained 3 days after the incident. A dose evaluation was carried out using these monitoring data to obtain the skin dose and the committed effective dose E(50) Sv.

# Practical measurement of site-specific dose coefficients for radon progeny dosimetry

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In the years since the release of ICRP publication 137 [1] there has been renewed interest in dose coefficients for radon progeny. ICRP publication 137 dispensed with the previous dose conversion convention in favour of dose conversion factors based on the ICRP respiratory system model. This led to significant change in long accepted dose conversion factors for inhalation of radon and its progeny and introduced the possibility of using a site-specific dose conversion factor. ARPANSA has developed equipment capable of measuring the aerosol data needed to determine site-specific radon dose conversion factors.

Two new pieces of monitoring equipment have been created – a radon progeny size spectrometer for the measurement of the activity size distribution of radon progeny; and a simple monitor for the direct measurement of dose conversion factors without the requirement to measure the detailed activity size distribution.

The radon dose conversion factors determined under a range of aerosol conditions, in both the laboratory and in the field, using this equipment will be discussed.

**References:**

[1] ICRP, Occupational intakes of radionuclides: Part 3. ICRP Publication 137. Ann. ICRP 46(3/4) (2017)

# Effective dose coefficients for radon and progeny: An update

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The International Commission on Radiological Protection (ICRP) publishes guidance on radiological protection against radon exposure in homes and workplaces. It has recently published dose coefficients for the inhalation of radon, thoron and their airborne progeny as well as recommendations for their use for the protection of workers [1]. Protection against radon is based primarily on measurement of air concentrations and optimisation. However, dose estimates are required for workers when radon levels are exceeded despite mitigation, or if the radon exposure is considered as occupational from the outset as in the case of mines. Dose coefficients also allow comparisons to be made of sources of public exposure.

The effective dose coefficient per unit exposure to radon progeny can be derived either by dosimetric calculations or by epidemiological comparisons. Taking account of both methods, ICRP have recently recommended a single rounded value of 3 mSv per mJ h $m^{-3}$ (approximately 10 mSv $WLM^{-1}$) to be used in most circumstances of radon exposure, for workers in buildings and in underground mines.

Recently, the United Nations Scientific Committee on the Effects of Atomic Radiation (UNSCEAR) conducted a review of radon epidemiology and dosimetry [2]. Lifetime risks were calculated by applying risk models to the updated Czech and Eldorado miner studies, to the newly published large WISMUT miner study, and to the combined 11 miner studies used in the BEIR VI report [2]. Given that the uncertainties from risk estimates are large, UNSCEAR concluded that its established dose coefficient of 1.6 mSv per mJ h $m^{-3}$ (5.7 mSv $WLM^{-1}$) should be retained for use in its comparisons of radiation exposures from different sources in a population.

This presentation explains and compares the reviews of the scientific evidence from UNSCEAR and ICRP. It also presents dose conversion factors based on recent published lifetime risk calculations. It is shown that the UNSCEAR and ICRP reviews are consistent and support the use of the ICRP reference dose coefficients for radiation protection purposes [3]. The UNSCEAR's established value is at the lower end of the range of calculated values whereas the ICRP reference dose coefficient is close to the central value. It is concluded that the ICRP dose coefficient should be used to calculate doses to workers.

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[2] UNSCEAR. Report of the United Nations Scientific Committee on the Effects of Atomic Radiation. *UNSCEAR 2019 Report, Annex B*, Lung cancer from exposure to radon. (2020).

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# ASSESSMENT OF THE IMPACT ON A CR39 NEUTRON DOSIMETRY SYSTEM OF THE NEW OPERATIONAL DOSE QUANTITIES PROPOSED IN THE ICRU REPORT 95

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In 2020, a new ICRU report on Operational Quantities for External Radiation Exposure was presented by ICRP. This document presents a new concept for definition of the operational quantities which is radically different to the previous ICRU concept. This change will have an impact on the metrology of the radiological protection units used to determine the risk relative to the ionizing radiation exposure of the worker. For instance, the conversion coefficients for determining the operational quantity in individual monitoring for neutrons at 0° may change by up to a factor of 2.85. (The name of the quantity will change from personal dose equivalent to personal dose.) The work presented assesses the impact on an individual dosimetry system for neutron exposure based on a CR39 technology. This impact is analyzed considering the criteria of the ISO 21909-1:2015

# EURADOS ISO/IEC 17025 GUIDANCE FOR IMS: SUGGESTIONS ON HOW TO INTERPRET AND IMPLEMENT THE REQUIREMENTS INCLUDING EXAMPLES FROM ACCREDITED LABORATORIES

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Individual monitoring of external radiation is an activity usually regulated by national regulatory bodies in most countries. Regulations generally contain technical requirements to be met by the individual monitoring services (IMS), in order to ensure that the measurements are correct and therefore the dosimetry results are reliable. In some countries, the requirements include or even consist of the accreditation of the service according to the standard ISO/IEC 17025: “General requirements for the competence of testing and calibration laboratories”. It is a fact that accreditation is a growing trend among European IMS as a way to guarantee confidence in their technical competence. The acceptance of the dosimetry results between countries and their indentation in the respective National Dose Registries is facilitated if laboratories conform to the ISO/IEC 17025 standard.
In the framework of the activities of EURADOS (European Radiation Dosimetry Group) working group 2 "Harmonization of Individual Monitoring in Europe" and attending to the concern of many European IMS in the process of accreditation, a guide has been prepared. The purpose was to assist and encourage IMS to apply for accreditation and to share the authors’ own experience with the process.
The guide intends to be a practical reference for IMS on how to interpret and implement the ISO/IEC 17025 requirements to the specific activity of a personal dosimetry service for external radiation, emphasizing those aspects of special interest. It includes examples from dosimetry laboratories already accredited. The major novelties from a new edition of ISO/IEC 17025:2017 are also identified in the guide. Finally, the guide aims to assist the auditing process, giving examples of auditor’s questions and how to show evidence of compliance.
The main findings will be presented.

# Uncertainty Estimation for the Dosimetry Service of Public Health Institute of Republic of Srpska using RADOS RE-2000 Semiautomatic TLD Reader


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Over the years ICRP, IAEA, EU and other organizations have published recommendations regarding uncertainty in dosimetry for exposed workers. Also IEC TR 62461, IAEA-GSG7 and EU RP 160 give specific guidelines for the application of uncertainty analysis in the field of monitoring of exposed workers, based on GUM framework.

The purpose of the study is to present the estimation of the uncertainty for the measurement of the personal dose equivalent Hp(10) using passive thermoluminescent detectors (LiF:Mg, Ti) placed in whole body dosemeters and measured with semiautomatic RADOS RE-2000 TLD reader used by Personal Dosimetry Laboratory of Public Health Institute of Republic of Srpska. The dosemeters are used for the estimation of the effective dose of about 1200 exposed workers.

The methodology used for evaluation of uncertainty, based on the above documents, consists of two stages: In the first stage, main sources of uncertainty in the measurements of personal dosimeters were identified (calibration factor, energy and angular dependence, non-linearity, homogeneity and reproducibility, fading and blank signal of the detector) and a model function that correlates the dose results with identified inputs was developed. At the second stage, the uncertainty was estimated in two ways: one analytical, based on the law of propagation of uncertainties and the central limit theorem (GUM framework) and the second, based on Monte Carlo techniques.

The results of analysis have shown that the estimation of uncertainty is realistic and the dose results are in compliance with the ICRP 60 recommendations on accuracy for radiation protection purposes.

Furthermore, a direct comparison of the analytical and the Monte Carlo technique is performed using the same input data. The results of uncertainties with the above two methods are comparable and it was shown that the use of Monte Carlo techniques can be a useful tool in order to validate the calculations based on the GUM methodology.

# The impact of radiation protection and other factors on workers' doses in industry and research using unsealed sources


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The purpose of this study was to show the trends in occupational exposure and describe the results of individual monitoring in the field of industry and research in Finland during years 1990–2020 in various workplaces using unsealed sources. In industry and research, the largest doses are received by researchers using unsealed sources and industrial tracer testing technicians. The aim was to investigate how different factors, such as introducing new protective shields, affect the radiation doses.

Dose data was collected from the Finnish national Dose Register. Both deep doses $H_p(10)$ and finger doses $H_p(0.07)$ were collected and analyzed to provide a comprehensive understanding of the field using unsealed sources. In addition, inspectors and other specialists were interviewed and reports of inspection were used. The impacts of factors, such as the use of new protective shields, economic situation, automatization and new radiation sources, were considered.

The common trend has been that the use of finger dosemeters has increased as the use of whole body dosemeters has decreased in both sectors. The collective finger dose in industry shows significant peaks around 2009 and 2014. As the use of fluorine-18 became more frequent, the finger doses increased distinctly. The collective deep doses in industry have stayed relatively low since 2006. Both in industry and in research, an increase in doses has always been followed by a decrease. Also, during the past few years, the trends in doses have been decreasing. As a conclusion, this shows that the improvements in radiation protection have been effective.

# Personal Dosimetry at the Paul Scherrer Institute

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The Paul Scherrer Institute (PSI) is the largest research institute for natural and engineering sciences in Switzerland. PSI develops, builds and operates complex large research facilities. Every year, more than 2400 scientists from Switzerland and around the world come to PSI to use the facilities to carry out experiments. Many areas at PSI are radiation protection areas. Depending on the radiation protection area, the work carried out and the time the users spend in these areas, they have to carry a personal dosemeter.

PSI runs an individual monitoring service in compliance with the Swiss legislation on radiological protection that is approved by the Swiss Federal Nuclear Safety Inspectorate. The service provides over 40000 dosemeters per year for internal and external customers consisting of whole body dosemeters for photons and neutrons as well as extremity dosemeters. A main part of dosimetry is performed especially in neutron dosimetry for external customers like CERN, DESY, Seibersdorf Labor GmbH and MPA-NRW.

This contribution gives an overview on the number of distributed dosemeters for internal and external customers, the classification of radiation workers at PSI and the according to that employed personal dosimetry techniques by the individual monitoring service and some statistics about the measured doses at PSI.

# A Model for Estimation of Internal Dose When Bioassay Analysis Decision Level is higher than the Derived Recording Level

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Monitoring of internal exposure to radionuclides is conducted in order to demonstrate compliance with regulatory requirements and to assess radiation risks from occupational exposure to workers. According to Israeli regulations, internal doses above a predetermined recording level should be detected and recorded. The assessment of internal dose is based on bioassay measurements, which might include *in vivo* measurements, such as whole body (WBC) and *in vitro* measurements, such as urine.

In many cases, bioassay analysis decision level is higher than the derived recording level and doses above the recording level might be undetected. A model was developed at the Nuclear Research Center Negev (NRCN) to estimate internal doses in such a case. The model assumes that bioassay measurements are log-normally distributed with a geometric standard deviation of 2.0. This assumption is based on the analysis of numerous measurements at the NRCN and on similar results that were reported in other researches. The assumption is applied to all the results that are recorded as below the decision level. An arithmetic mean of an assumed log-normal distribution, in which 95% of the distribution is contained below the measurement decision level, is determined. The intake and dose are then evaluated from the reconstructed mean by assuming a constant chronic intake regime between consecutive bioassay measurements.

# Occupational radiation dose to the hand of medical staff who assist in diagnostic computed tomography

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**Background**: Chronic radiation exposure increases the risk of skin damage in medical personnel engaged in fluoroscopy [1,2]. However, no reports have evaluated the hand dose measurement in personnel assisting adult patients undergoing high-dose computed tomography (CT) imaging procedures.

**Purpose**: We investigated the occupational radiation dose to the hand in personnel assisting patients during CT scans for diagnostic purposes to evaluate the compliance with the equivalent dose limit for the hand (500 mSv/year).

**Method**: The occupational doses of personnel (n = 9; 6 intensivists, 3 radiological technologists) assisting patients during a CT scan were measured (n = 89, April 2017 - May 2018). We installed a radiophotoluminescence glass dosimeter (RPLD; GD-302M) on the dorsal aspect of both hands to measure the dose to the hand. The energy calibration constant of the GD-302M is 0.34. The working practices for the personnel included providing respiratory assistance using a back-valve mask, holding the patient's head by hand (head holding), and observing the patient in the CT room. The differences in occupational radiation dose among different CT-assisting methods were confirmed using the Kruskal–Wallis one-way analysis of variance. If the analysis result was significant; the difference between the methods was evaluated using the Dunn test (with Bonferroni correction). Statistical significance was set at $p < .05$ (SPSS, Ver. 25.0).

**Result**: The occupational radiation dose to the right hand (median [confidence interval]) was significantly higher for head holding (Head holding: 1,047 [244–1,374] μGy, back-valve mask: 254 [173–408] μGy, observing: 40 [23–98] μGy; $p$ <0.01). Similarly, the occupational dose on the left hand was also significantly higher for head holding (985 [221–1,129] μGy, back-valve mask: 281 [223–358] μGy, observing: 52 [19–140] μGy; $p$ <0.01).

**Conclusion**: We found that the radiation dose to the hand of CT-assisting personnel was highest with head holding (right: 1,047 μGy, left: 985 μGy). In addition, on evaluating the occupational dose (median) to the hand measured during head-hold, it was found that this action can be performed 500 times without exceeding the annual equivalent radiation dose limit for skin (500 mSv/year).

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# SORGENTINA-RF nuclear fusion device: preliminary study of the shielding allowing workers annual dose limit compliance

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In the framework of an agreement among ENEA, Emilia-Romagna and Toscana Regions an accelerator-driven 14 MeV neutron source of new concept, denominated SORGENTINA-RF, will be installed in Brasimone ENEA Research Centre, near Bologna, to test the feasibility of producing radionuclides of medical relevance using fusion neutrons [1].
In particular, the main goal of the facility is generating $^{99}$Mo, a precursor of $^{99m}$Tc, a radionuclide widely used in nuclear medicine examination [2], using the 14 MeV fusion neutrons produced by the plant. These neutrons will irradiate a metallic sample containing $^{100}$Mo inducing $^{100}$Mo(n,2n)$^{99}$Mo reactions.
The intensity of the designed source is foreseen to be about $7.10^{13}$ $s^{-1}$. The facility will be installed in a building of the Brasimone Research Centre and a proper radiation shielding structure will be required to workers annual dose limit compliance and to prevent undue exposure.
The present work describes the study performed for the design of a proper shielding structure design, that was aimed at complying with the 0.01 mSv/h dose rate limit proposed for the external surface of the shielding itself during beam-on operations.
Monte Carlo simulations with a multiprocessor version of MCNP6 code [3] and the CRESCO-4 grid infrastructure [4] have been employed to define the shielding structure and evaluate the ambient dose equivalent rate in some selected points of the facility.
The simulations have been performed using an isotropic 14 MeV neutron source and neglecting the real target structure and the ancillary equipment that will be installed inside the shielded compartment. This approach is quite conservative since the angular/energy distribution of the emitted neutrons and the self-shielding effect due to the Sorgentina-RF structural components are not taken into account. The obtained configuration consists of a “full shielding” (floor, walls and ceiling) structure of 2 m standard concrete followed and 1 m baritic concrete, the latter aimed at the attenuation of the intense secondary photon field induced by the absorption of the neutrons in the concrete. The access to the internal compartment is provided through a maze, closed by a properly shielded door.

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# CIEMAT PARTICIPATION IN SIX EURADOS INTERCOMPARISONS FOR WHOLE BODY DOSEMETERS IN PHOTON AND BETA FIELDS (2008-2018)

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The participation of individual monitoring services in intercomparisons is an excellent way to demonstrate the reliability of their results. If the service is accredited, this participation is mandatory for each accredited testing method.
The Ciemat External Personal Dosimetry Service (EDS) has implemented a quality system based on the ISO-17025 [1] standard, which requires the participation in intercomparisons in a planned manner. Since 2008, the Ciemat EDS has participated in the six international intercomparison exercises of whole body dosemeters in photon and beta radiation fields organized by the EURADOS Working Group 2: "Harmonization of individual monitoring". The radiation qualities tested allow the participating IMS to obtain information on the energy response to photons and mixed radiations, the angular response and the linearity of the dosimetry system.
The whole body dosemeter at the Ciemat EDS consists of two $Li_2B_4O_7$ detectors and two $CaSO_4$ detectors under different filtrations (Panasonic UD-802 model). The dose assessment is performed using a dose calculation algorithm, based on the ratio between elements, which reports the values of Hp(10) and Hp(0.07) as the final result.
This paper presents the conclusions of the analysis of the results obtained by the Ciemat EDS in the participation in the six intercomparisons previously mentioned. All results were included within the acceptance criteria of the ISO-14146 [2] standard (also known as “trumpet curves”), so participation is considered satisfactory. The temporal evolution of characteristics such as linearity, angular response and response in mixed fields are also analyzed as additional information on the stability of the dosimetric system.

**References:**

[1] ISO 17025:2017, Nuclear Energy -- Radiation Protection – General requirements for the competence of testing and calibration laboratories
[2] ISO 14146:2018: Radiological Protection - Criteria and performance limits for the periodic evaluation of dosimetry services.

# CIEMAT PARTICIPATION FOR EXTREMITY DOSIMETRY SYSTEMS IN EURADOS INTERCOMPARISONS (2009-2019)

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The extremity dosemeter of the Ciemat External Dosimetry Service (EDS) consist of one $^{7}Li_2{}^{11}B_4O_7$:Cu detector (model UD-807 from Panasonic) inside two types of ring-holders with different filtration. The type of holder is selected depending on the presence or not of low energy beta radiation in the user facility.
This paper deals with the summarized results that were obtained by the Ciemat-EDS extremity dosimetry system in the intercomparison exercises organized by EURADOS during the last 10 years. Participation has provided compliance with a relevant requirement for the accreditation, according to the standard ISO-17025 [1]. The intercomparison exercises were performed in 2009 and 2015, and now there is another in progress during 2019 whose irradiations are being performed.

- The Intercomparison exercises were coordinated and evaluated by the WG2 “Harmonization of individual monitoring” of EURADOS.
- Reference irradiations were carried out by metrology accredited laboratories.
- Readouts were performed using two Panasonic readers, models UD-710 and UD-716, calibrated by accredited metrology laboratories in terms of the quantity Personal Dose Equivalent Hp(0.07) in hands.

The results show that gamma and the most beta irradiations comply with the acceptance criteria of the ISO-14146 [2] (also known as trumpet curves). Only a very few outliers were produced, mainly related to low beta energies (Kr-85). Due to anomalous results by Kr-85 source, and punctually due to irradition Sr-90 / Y-90 at 60º in 2015, two deviations in the Quality System were managed, concluding that the ring-holder should be selected depending on the beta emitter energy, and moreover, light protection should be careful during the preparation before dosemeters read out. Analysis of Eurados IC2019 will be also presented, if available.
The document concludes that the established criteria have been satisfactorily accomplished, being suitable for dosimetric surveillance of exposed workers.

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[1] ISO 17025:2017, Nuclear Energy -- Radiation Protection – General requirements for the competence of testing and calibration laboratories
[2] ISO 14146:2018: Radiological Protection - Criteria and performance limits for the periodic evaluation of dosimetry services.

# Calculation of conversion factors between dose-length products in CT and dose-area products in CBCT for common imaging tasks for head and trunk

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Cone-beam computed tomography (CBCT) has emerged as an alternative to computed tomography (CT) for many clinical applications. However, comparison of the doses of the two methods is difficult, as the application-specific dose quantities in CT and CBCT are inherently different: the dose-length product (DLP) used in CT is a quantity related to the average dose in a PMMA phantom, the so-called CTDI phantom. The dose-area product (DAP) used in CBCT is measured free-in-air as the product of the incident air kerma and the collimated area. To date, it has been difficult or impossible to compare the technical quantities involved, and specifically the risk to a patient that results from exposure. The aim of this work was to find conversion factors that will allow common diagnostic reference levels (DRL) to be established for identical imaging tasks.
In our investigation, we followed a risk-based approach. In CT imaging, the x-ray tubes make full rotations, and, thus, patients are exposed from all directions (360°). In CBCT imaging, most units follow a trajectory of less than 360°. Thus, the distribution of doses to a patient's organs is different, even for similar average total doses. Consequently, we used organ doses and the resulting effective dose $E$ to enable a comparison. The effective dose $E$ was calculated from simulation studies together with the DLP for CT and DAP for CBCT, from which conversion factors $k_{CT} = E_{CT}/\mathrm{DLP}$ and $k_{CBCT} = E_{DVT}/\mathrm{DAP}$ were derived. Then, a risk-based conversion factor was found by comparing scans with identical effective doses, i.e., $E_{CT} = E_{DVT}$. Under the assumption that $k_{CT}$ and $k_{CBCT}$ stay constant when the effective doses are varied, the conversion can be expressed as $\mathrm{DAP} = (k_{CT}/k_{CBCT})*\mathrm{DLP}$; thus, the risk-based conversion factor can be expressed as $k = k_{CT}/k_{CBCT}$.
Input parameters were obtained from measurements on real-world CT and CBCT units. The simulations were performed using version 1.6.1 of ImpactMC, a Monte Carlo simulation program, on the male reference phantom from ICRP report 110, and by applying organ dose weighting factors from ICRP report 103. Eight scan regions in different parts of the head and trunk were chosen and the dose quantities calculated for eight CBCT and two CT units. The spectra and collimation were extracted from the measurements. The risk-based conversion factors $k$ were calculated for different body parts equalling the effective doses for corresponding studies in CT and CBCT imaging.
Conversion factors $k$ of 60(15) cm, 45(14) cm, 53(18) cm, and 41(10) cm were found for the four head scan regions and 119(40) cm, 117(30) cm, 81(17) cm, and 124(38) cm for the four trunk scan regions under inspection.
Using the $k$-factors, one can calculate the corresponding technical dose quantities of CBCT and CT for specific imaging tasks. This enables a specification of DRL for corresponding imaging tasks in CT and CBCT that lead to the same risk.

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# Combined measurement techniques for the investigation of radiation exposure in ultrashort pulsed laser machines

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Performing radiation protection tests in ultrashort pulsed laser (USPL) machines is a challenging task since the generation of laser-induced X-rays is an unstable process and depends on many variables including laser parameters and manufacturing processes. USPL systems operating in the femto- to picoseconds range made high precision in materials micromachining possible. Unique advantages of USPLs are fast and localized energy deposition, nearly melt free ablation and minimal thermal load on the workpiece [1, 2]. Moreover, the high average output powers (>10 W) and high repetition rates (from kHz to MHz) available in the nowadays USPL machines fulfil industrial requirements in terms of throughput and productivity, opening a wider range of applications. However, due to the high laser peak intensities reaching above $10^{13}$ W/cm$^2$ during USPL processes, undesirable laser-induced X-ray emissions can be generated [3, 4, 5, 6]. Up to now, very limited number of investigations have been made worldwide to address this radiation protection issue. There is a need for understanding and characterizing such complex laser-matter interactions that can lead to ionizing radiation, posing a potential health hazard to the personnel working in the vicinity of the USPL machines. In this work we present an experimental approach that combines 2-D radiography imaging, photon-energy spectroscopy and dosimetric measurements to monitor and quantify laser-induced X-ray emissions during USPL operations. Our aim is to investigate different laser and processing parameters regarding the radiation exposure and energy distribution of the laser-induced X-rays, as a reference for radiation protection tests. Our recommendations are based on empirical results from investigations carried out on an USPL machine with a 20 W laser. The data were measured using constant pulse duration (274 fs) and laser spot diameter (33 µm) at 1030 nm wavelength, while the repetition rate and pulse energy were variated from 50 to 400 kHz and 371 to 46 µJ, respectively. Such parameters resulted on laser average intensities from $3.9 \times 10^{13}$ to $3.2 \times 10^{14}$ W/cm$^2$ on tungsten targets in the shapes of flat plates and rods.

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# In-field radon intercomparisons in Italy: the experience of three editions organized by AIRP

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The periodical participation at inter-laboratory comparisons is a tool that laboratories may use to assess or to monitor their own performances. Typically, radon intercomparison exercises are carried out in radon reference chambers under stable and controlled conditions in term of radon levels, temperature, humidity and atmospheric pressure. Nevertheless, radon laboratories are interested in testing their monitoring systems by in-field exercises, taking place in less controlled and more variable real conditions, like ones in which radon passive devices are usually exposed.
Up to now, three editions of the International radon in-field intercomparison for passive measurement devices were promoted by Italian Radiation Protection Association (AIRP) and were held in Italy in 2014 (two radon level exposure tests) [1, 2], in 2016 (three radon level exposure tests) [3] and in 2019 (five radon level exposure tests).
The focus of the first edition was the evaluation of radon concentration in severe environmental conditions, while for the second edition the evaluation of the effect of long exposure time and the simultaneous presence of high level of radon and thoron were investigated. For the third edition, the focus was the evaluation of factors potentially affecting the quality of radon measurements in real situations (i.e. in absence of transits, long exposure time at low and high radon levels, thoron interference).
The increasing number of participants confirm the interest of radon laboratories for in-field proficiency tests: the laboratories registered for the different editions were 46, 49 and 79 respectively.
According to the ISO 17043:2010 [4], for each radon level exposure test, the percent difference (*PD*), the normalized error ($E_n$) and the z-score have been calculated for each device set, in addition to standard statistics - e.g. arithmetic mean, median, standard deviation and coefficient of variation.
In this work, the authors present a synthesis of the results obtained in the three radon intercomparison exercises, highlighting the lesson learnt in relation with the focus of each one. Furthermore, the overall results obtained by laboratories that have participated in several editions will be discussed.

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# Investigation of neutron stray fields close to the two-meson targets of the PSI proton accelerator facility

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The high intensity proton accelerator facility (HIPA) at the Paul Scherrer Institute (PSI) accelerates protons to an energy of 590 MeV with currents up to 2.4 mA, i.e. 1.4 MW beam power. The beam feeds four main experiments using individual targets. Two of these targets are rotating wheels made of different sizes of carbon producing mesons used for particle physics and material research [1]. The areas adjacent to these targets are heavily shielded by several meters of iron and concrete.
The neutron stray field at different positions outside the shielding close to these targets has been investigated using the PSI extended range Bonner sphere spectrometer (ERBSS). It consists of 10 moderator spheres made of polyethylene and 4 spheres modified with metal shells, enhancing the sensitivity for neutrons with energies above 20 MeV. The obtained data was normalized to the proton current indicated by a resonance chamber upstream the first target. Two commercially available survey instruments for neutron and photon radiation constantly monitored the stability of the field during the measurements. The spectral neutron distribution was determined by applying Bayesian methods [2], which were adapted for measurements of neutron stray fields behind shielding at high energy accelerators.
The measurements within restricted access areas resulted in ambient dose equivalent rates from 25-50 µSv/(h mA) with significant contribution of high-energy neutrons.
The comparison of doses indicated by a commercially available survey instrument suitable for measurements in fields with a high-energy neutron component showed a reasonable agreement with the dose values obtained from the ERBSS measurement. However, it is desirable to apply in-field calibration factors derived from spectrum measurements to reduce the uncertainty of dose values obtained with survey instruments.

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# A method to identify and localize a hotspot in the lungs using a whole body counter for improved estimate of the deposited activity

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Direct, in vivo measurements are routinely performed to identify an unexpected intake and as part of the post-exposure incident evaluation process to determine the quantity of radioactive material deposited in the respiratory tract. Measurements of the deposition, retention and removal rate of the inhaled material are important parameters for calculating the dose to the lungs [1-2]. However, it is conventional practice in incident evaluation to adopt generic, default values for these parameters along with the assumption that the inhaled aerosols are homogeneously distributed in the tissue to expedite reporting an estimate of intake and dose for regulatory compliance and as guidance to initiate remedial action [3].

A new method has been developed to identify and localize a single hot particle in the lungs using an array of four high purity germanium (HPGe) detectors [4]. The method is based upon calculating a sequence of count rate ratios (generated by each individual detector in the array) that are combined to uniquely associate the deposition to a hot particle, rather than the default assumption of a uniform activity distribution. Identification and localization of the hot particle is determined from a single in vivo measurement in which detectors are positioned above and below the thorax. The method was tested using an anthropomorphic thorax phantom in which point sources of $^{241}$Am, $^{137}$Cs and $^{60}$Co were individually inserted in the lungs at 15 different locations and measured using a scanning bed whole-body counter. Depending upon source location and photon energy, a bias of -35% up to +76% could be introduced by falsely assuming a uniform activity distribution in the lungs. This bias would directly translate to an erroneous dose estimate to the lungs.

By employing the method one can (a) differentiate a single hot particle in the lungs from a homogeneous distribution of activity, (b) localize the hot particle, and (c) improve the accuracy of the measured activity of the hot particle by using a more appropriate detection efficiency. It was demonstrated that by using the appropriate detector efficiencies for the single hot particle, the bias associated with the activity determination is reduced to below 10% and ~2% in average.

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# Investigation on applicability of small radiophotoluminescence dosemeter to extremity dose monitoring for medical personnel


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An additional individual monitoring using a ring badge must be appropriately conducted, when inhomogeneous exposure around radiation workers' extremity were critical for radiation workers. Various types of ring badges consisting of such as thermoluminescent dosemeters (TLDs), optically stimulated luminescent dosemeters(OSLDs) are widespread and available for routine monitoring, particularly in the medical and nuclear fields. In addition, the ring badge applying new materials has been proactively developed [1].

We have investigated a practical and effective eye lens dose monitoring of a commercially available radiophotoluminescence dosemeter (RPLD) with small dimensions for medical personnel [2],[3]. The RPLD, the GD-302M and the GD-352M (AGC Techno Glass, Shizuoka, Japan) have a dimension of 13.0 mm in length and 2.8 mm in diameter for outer plastic casing and will also be a promising candidate for additional monitoring of the extremities of medical personnel. We investigated the applicability of this commercially available RPLDs to extremity dose monitoring by performing a characterization of the dosemeters by the experiments and the Monte Carlo (MC) calculations. The preliminary MC calculations for the energy dependence of the GD-302M without tin compensation filter revealed that the GD-302M overestimates the personal dose equivalent, $H_p(0.07)$ by a factor of more than three in energy range between 20 and 80 keV. On the other hand, the relative energy response of the GD-352M to the $^{137}$Cs gamma rays (662 keV) showed satisfactory to the criteria of the personal dosemeter for the extremity dosimetry i.e., from 0.71 to 1.67 with energies from 30 keV to 1250 keV.

The presentation will show the detailed investigation of the characterization of the RPLDs as an extremity dosemeter and discuss the applicability of the RPLDs to additional monitoring of the extremities.

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# Revision of the ISO 21909 standard on the passive personal neutron dosemeters

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In 2012, ISO TC 85/SC 2 Working Group 19 dealing with "Individual monitoring for external radiations" conducted important work revising the ISO 21909 standard on passive personal neutron dosemeters. There was consensus in the community that the 2005 version [1] needed to be revised for two reasons. It was first essential to achieve a more universal document. The performance tests and criteria needed to be described in order to lead similar results whatever the considered techniques are. Moreover, the document had to better reach harmonization between performance tests and conditions of use at the workplaces.
The document was completely re-written to be constraining enough to ensure that any neutron dosimetry system fulfilling the criteria of this new document would be reliable in most of the usual work situations, in terms of dose level energy and direction distribution of the neutron fluence.
Part 1, published in 2015 [2] and with a first revision to be available end of 2021 to consider first feedbacks from IMSs, provides performance and test requirements for the measurement of personal dose equivalent, $H_p(10)$, for neutrons ranging from thermal energy to approximately 20 MeV. Part 2, to be available early 2022 [3], is dedicated to systems which do not meet the criteria with regards to the energy and directional dependence of response described in part 1. Part 2 will provide methodology and criteria to qualify the dosimetry system at workplaces where it is used.
This paper explains the reasons behind the revision and presents the main evolutions induced by the revision. A specific focus on the philosophy of this new document will be presented so Individual Monitoring Services may better understand how to apply the requirements in practice.

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[1] ISO 21909:2005, Passive neutron dosimetry systems - Performance and test requirements (2005)
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[3] ISO 21909-2, Passive neutron dosimetry systems - Part 2: Methodology and criteria for the qualification of personal dosimetry systems in workplaces, to be published

# Summary of performance testing requirements for environmental dosemeters compared to those used in the workplace


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Outdoor active (area)dosemeters are becoming increasingly important for environmental monitoring, notably in early warning and non-governmental networks that have emerged because of increasing awareness of potential nuclear accidents. A clear separation of workplace and environmental area dosemeters is absent. This has led to a state whereby different international standards for area environmental and area workplace dosemeters are being used. These non-harmonised type test requirements set in the different international standards for environmental and area dosemeters are related to the same operational radiation protection quantity - ambient dose equivalent rate. Even though most requirements for area workplace and environmental dosemeters are similar, differences are needed, for example the dose rate range and the angular dependence.
Therefore, harmonisation of the type test requirements for both environmental and area workplace dosemeters will improve the type testing procedures and standard acceptability criteria for these dosemeters, impacting on metrology calibration and testing laboratories, dosemeter manufacturers, stakeholders and end-users.
One of the main objectives of the 17RPT01 DOSEtrace (Improving Europe's calibration facilities for radiation monitoring equipment) project is the development and validation of traceable measurement capabilities for operational radiation protection quantities. Harmonisation of type test requirements for the measuring instruments which measure ionising radiation dose in terms of ambient dose equivalent rate is closely related.
In the presented work, existing type test requirements set by different international standards were summarized and the most relevant requirements for both workplace and environmental area dosemeters were extracted. Two sets of requirements were formulated – one for workplace area monitors and one for environmental area monitors. The requirements and test methods were experimentally tested with selected area dose rate meters.
The results could be used as an input for the IEC standardisation body to set up reliable unified type test requirements for both dosemeter types.



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# Evaluation of the Quality Assurance Programme for PADC Dosimetry Plastic

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UK Health Security Agency, UKHSA (formerly Public Health England) operates a neutron personal dosimetry service using poly-allyl-diglycol-carbonate (PADC, also known by trade name CR-39®) dosemeters that detect thermal, epithermal and fast neutrons. The dosemeter is designed to measure doses from neutrons to the whole body and to the skin in terms of the radiation quantities $H_p(10)$ and $H_p(0.07)$ as required by the UK Health & Safety Executive (HSE).

The service relies on a thorough quality management system to validate the PADC quality prior to use and identify and reject unsuitable material. Detector elements used in the dosemeter are cut from larger parent sheets, the characteristics of which are assessed on delivery by taking quality acceptance (QA) samples. The samples are assessed to calculate the mean sheet thickness, mean intrinsic background pit count B, standard deviation on the same $\sigma_B$, and mean sensitivity S. From these parameters the decision threshold ($L_C$) is calculated as $L_C = 2\sigma_B/S$, and the maximum dose on a background detector, $H_{max}$, above the underlying background, is calculated. For acceptance, sheets must possess values of $L_C$ and $H_{max}$ which lie below 0.15 mSv, and the coefficient of variation on S must be 15% or less. Of the 90 detectors on a single PADC sheet, the QA samples comprise 10 detectors taken for background evaluation and 8 detectors for sensitivity evaluation.

The ability of the QA process to correctly identify PADC sheets suitable for customer use can be tested by processing all the remaining detectors on a sheet left after the QA samples are removed. This provides data from nearly the whole sheet rather than just the restricted QA sample. This was done for selected sheets, which had their remaining detectors processed without being dosed (etched as backgrounds), to obtain the true underlying background pit distribution for the whole sheet. The whole-sheet data was then sampled, each sample being 10 detectors just as for QA, using a sampling program that returned 3000 samples. The results for each sample were analysed as a QA sample would be. This provided information on whether the QA pass/fail result was representative of the underlying pass/fail for the whole sheet, and whether a sheet was correctly passed or failed.

By assessing a variety of failed sheets, the QA methodology was reviewed to consider whether improvements to the current criteria are required.

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# The European Metrology Network for Radiation Protection: Development of a joint and sustainable metrology infrastructure

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European Metrology Networks (EMNs) are of strategic importance for the European Association of National Metrology Institutes (EURAMET) in order to realize the vision of EURAMET and its members to have world-leading metrology capacities based on a high-quality scientific research infrastructure.
The development of an EMN in the field of radiation protection and associated metrology was planned by the consortium of the EMPIR project 19NET03 supportBSS and the EMN for Radiation Protection was officially approved on September 16, 2021.
It will interact with innovative technological developments and is intended to be the central contact point between the metrological communities and the relevant radiation protection stakeholders, including regulatory authorities, standardization bodies, manufacturers, users of radiation sources as well as international organizations and platforms dealing with radiation protection such as HERCA, BIPM, IAEA and EURADOS to meet their needs and interests.
One of the central challenges in achieving these goals is the establishment of a joint and sustainable European metrology infrastructure that underpins radiation protection regulation. In this work we present the development of an initial plan that takes such infrastructure development into account.

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# PERSONAL DOSIMETRY FOR POSITRON EMITTERS, AND OCCUPATIONAL EXPOSURES FROM CLINICAL USE OF Ga-68

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**AIMS**: The current status and issues regarding positron dosimetry during PET-CT nuclear medicine are briefly summarized. The suitability for positron dosimetry of the UKHSA extremity and eye beta-gamma personal thermoluminescence dosemeters (TLDs) are then determined. Finally, doses to workers from clinical gallium-68 exposures are explored, in both highly idealized and hospital environments.
**METHODS**: Monte Carlo modelling is performed to determine dosemeter responses to positron sources under realistic exposure conditions, and in turn derive appropriate sets of calibration factors. Results for carbon-11, nitrogen-13, oxygen-15, fluorine-18 and gallium-68 are presented, all of which are commonly used in PET procedures, along with the $H_p(0.07)$ and $H_p(3)$ conversion coefficients for those sources. An idealized set of measurements that represent clinically relevant $^{68}$Ga-DOTATATE exposure scenarios are then described, including reproducible mock-ups of individuals manipulating vials and syringes containing the radionuclide solution. Extremity, eye and whole-body doses are considered, with the latter supported by additional Monte Carlo modelling of an anthropomorphic voxel phantom. A case-study is then presented that explores occupational doses during routine use of $^{68}$Ga within a hospital environment over a period of weeks.
**RESULTS**: The existing designs of dosemeter were found adequate for assessing exposures to PET radionuclides, with only relatively small and acceptable losses of accuracy incurred, even if their routine calibrations to $^{137}$Cs are maintained. In the controlled exposures to $^{68}$Ga-DOTATATE, positron dose equivalent rates per activity were seen to be broadly similar to previous values, though published data are scarce. Some of the dose rates were quite high, and the extremity dosemeter results demonstrated significant variations that were dependent on the precise exposure conditions (e.g. task, dosemeter location, etc.). Eye and body dose rates per activity were found to be lower than for localized skin exposures, though such exposures may be more prolonged. Some of the doses found during the hospital measurements were also large, though all were within occupational dose limits when extrapolated annually. In general, however, a wide range of results was again evidenced.
**OUTCOMES**: In lieu of ICRP / ICRU recommendations for positron dosimetry, the overall conclusion that PET-CT workers may be monitored using existing dosimetry resources is important. However, a significant outcome from the measurements is that the positron doses recorded by extremity dosemeters may underestimate maximum doses to individuals: both depend on the precise exposure conditions, which can differ due to wear-location, the procedure, or during different stages of the lifetime of the $^{68}$Ga-generator, for example. The present work extends the limited data currently available on occupational doses from $^{68}$Ga, whilst reinforcing the need for accurate personal monitoring of the body, eye, and both hands of individuals working with it.

# Type testing of the ciemat extremity dosemetry system for $H_p(0.07)$ assessment

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The extremity dosemeter of the Ciemat External Dosimetry Service (EDS) consist of one $^{7}Li_2{}^{11}B_4O_7$:Cu detector (model UD-807 from Panasonic) inside two types of ring-holders with different filtration. The type of holder is selected depending on the presence or not of low energy beta radiation in the user facility. Results are provided in terms of the magnitude Personal Dose Equivalent Hp(0.07) in hands.

This paper deals with the tests carried out to complete the extremity dosimetry system type-testing, according to the "Spanish Protocol for characterization and calibration of extremity dosemeters", developed by the Spanish regulatory body, and based on the withdrawn ISO 12794:2000 [1] standard, whose scope was covered by IEC 62387 in 2012 [2] and later editions.

The following performance requirements were tested: homogeneity, reproducibility, linearity, energy and angular response to photon and beta radiations, stability under different environmental conditions, detection threshold, residue, self-irradiation and effect of light exposure on the dosemeter.

- Reference irradiations were carried out by the Ciemat Ionizing Radiations Metrology Laboratory.
- Rest of required irradiations was done in the Ciemat EDS panoramic gamma irradiator J.L. Shepherd, model 142-S.
- Readouts were performed using two Panasonic readers, models UD-710 and UD-716, calibrated in terms of the operational quantity Hp(0.07).

The results show that most performance requirements comply with the Spanish protocol acceptance criteria. Only small deviations were detected, mainly related to environmental conditions, dosimeter residue and optical fading. The paper concludes that the Ciemat EDS is suitable for the dosimetric surveillance of exposed workers, according to the Spanish regulations.

**References:**

[1] ISO 12794:2000, Nuclear energy — Radiation protection — Individual thermoluminescence dosemeters for extremities and eyes

[2] IEC 62387:2012: Radiation protection instrumentation - Passive integrating dosimetry systems for personal and environmental monitoring of photon and beta radiation.

# Simulation and characterization of radiation fields on the LRI Facility

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The Ionizing Radiation Laboratory (LRI, acronym in Portuguese) facility is designed to provide gamma and neutron radiation test fields, allowing scientific research on devices, components and materials radiation effects of interest to the Brazilian aerospace industry. The radiation facility consists of a $^{60}$Co source and a DT (deuterium-tritium) neutron generator.
The dose rate for gamma radiation ranges from 1.3 Gy/h to 50 Gy/h (Si), values that reach the standard and low dose rate windows established by the ESA/ESCC 22900 Standard [1] for electronic devices TID (Total Ionizing Dose) testing. Field calibration was performed using an ionization chamber traced to the national metrological standard.
The neutron fluence of the neutron field was characterized by a secondary standard "Long Counter" type detector, previously traced to the national metrological standard [2] and it were also compared with measurements obtained through thermoluminescent dosimeters [3]. The neutron field characterization inside the experimental bunker was determined through a computational simulation by the Monte Carlo method, using the MCNP code for different positions and distances from the DT neutron generator.
The total neutron production rate in 4pi geometry was $2 \times 10^8$ n/s for this test condition. The conversion factor for the neutron production rate, based on a reference detector counts that is used for beam tracking, was 1073734 +- 35620 n/count. The fluence, spectrum and angular distribution of the field as a function of distance was detailed and compared with the experimental results.

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# Calibration of thyroid counter system with various neck phantoms

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The in vivo measurement of radioiodine activity gathered in thyroid gland is one of the most used techniques for routine internal radiation exposure monitoring. This method allows to register energy spectrum of gamma radiation emitted from thyroid gland and calculate iodine activity. To get the correct evaluation the efficiency calibration is needed. The calibration may be performed with physical radiation reference sources placed in the special neck phantoms or with numerical methods.

This work presents the results of comparing efficiency calibration of Thyroid Counter System routinely used in Radiation Protection Measurements Laboratory NCBJ. The calibration was performed with various neck phantoms – water phantom developed at Radiation Protection Measurements Laboratory [1], SCK•CEN thyroid phantom [2], ANSI neck phantom [2] and also using numerical Monte Carlo methods with MCNP/X code.

The calibrations results were compared, the differences on assessment of the iodine activity in thyroid for various neck phantoms were calculated.

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# Spanish sixth intercomparison among external individual monitoring services

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**Introduction:** The Spanish Nuclear Safety Council (CSN) organizes regular dosemeter intercomparisons among Individual Monitoring Services (IMSs). In 2020, an intercomparison for whole body dosemeters in photon fields was carried out. A total of 21 IMSs participated in the study. The main objective was to verify the performance of the dosimetry systems and their compliance with the requirements of the Spanish regulator.

**Methodology:** Irradiations were performed at the secondary standard Calibration Laboratory at the Universitat Politècnica de Catalunya (UPC), accredited to EN ISO/IEC 17025 (2015) by the Spanish Accreditation Body, ENAC. The irradiation plan was designed based of the recommendations of ISO 14146 (2000). Participants were informed that dosemeters would be irradiated with photon irradiation qualities from ISO 4037-1 (2019), following ISO 4037-3 (2019) calibration procedures. In total, eight irradiation conditions were selected, including normal and angular incidence of the radiation, together with one mixed field. The performance limits according to ISO 14146, commonly known as “trumpet curves”, were considered in the results evaluation. The standard ISO 14146 allows a maximum of one-tenth of the dosemeters irradiated to exceed these limits. The response (R) was calculated for each dosemeter by dividing the participant's result by the reference dose values (given by UPC).

**Results:** All participants fulfilled the ISO 14146 requirements (max. 3 outliers). 18 out of 21 IMSs provided all response values within the limits. Two services presented difficulties to asses $H_p(10)$ for low X-ray qualities, with 1 and 3 outliers, respectively. Finally, 1 service presented 2 outliers with a response higher than 3 and associated to a problem with the readout of the two dosemeters.

**Conclusions:** Despite the situation caused by the COVID-19 pandemic, the 2020 intercomparison could be carried out as planned. The overall results are considered very satisfactory and highlight an excellent performance of the services. Compared with the previous Spanish intercomparison organized in 2013, the number of outliers has decreased significantly from 4.5% to 0.6%. In particular, the improvements were found for the response at low X-ray energies.

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# Project CIEMAT-CSN: Personal Neutron Dosimetry (DOPEN)

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## AIMS

Photonic and beta field dosimetry is a fairly well resolved issue, with effective measurement techniques and specific regulations to prove the reliability of the results.

However, neutron dosimetry presents considerable technical difficulties, so there is no system or method that meets all the measurement requirements in the different neutron fields found in practice, especially for personal dosimetry.

This communication presents the project "DOPEN: Personal Neutron Dosimetry", signed in 2020 by the Ciemat and the Spanish Nuclear Safety Council (CSN), aimed to the development of a personal neutron dosimetry system, using trace dosimeters based on PADC (poly-allyl-diglycol carbonate) plastic detectors, according to the applicable international standards [1].

## METHODS

CIEMAT has the TASLImage equipment, manufactured by the English company Track Analysis Systems, Ltd., which is a complete image analysis system for scanning and analyzing plastic PADC detectors.

The project, which will last 4 years, is structured in three phases:

- PHASE 1: Design and characterization of the dosimeter. Acquisition of the necessary instrumentation and equipment.
- PHASE 2: Optimization of the technique with advice from other European laboratories that routinely use similar techniques.
- PHASE 3: Calibration and dosimetry characterization of the system according to international recommendations.

## EXPECTED RESULTS

The neutron personal dosimetry system resulting from the project will be useful in all operations in which a significant contribution of the neutron dose is expected, and in particular for:

- Spanish nuclear power plants,
- the start-up of the ATC (Centralized Temporary Warehouse),
- the proton therapy facilities, operational cyclotrons, PET (positron emission tomography) and accelerators for radiotherapy that exist in Spain,
- the facilities such as CIEMAT itself, where the Neutron Standards Laboratory is located,
- Radiological emergencies with exposure to neutron fields.

## CONCLUSIONS

With this joint Ciemat-CSN project it is expected to improve the Spanish measurement capacity for mixed neutron and gamma radiation fields and to increase the reliability of dosimetric control methods for exposed workers to these radiation fields, in line with what results in a common practice in other European countries.

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# Problems in the use of LiF:Mg,Cu,P (TLD-100H) detectors for individual monitoring

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Individual and environmental monitoring benefited considerably with the introduction of the LiF:Mg,Cu,P (TLD-100H) detector due to its hypersensitive properties. However, its use also comes with some drawbacks. This paper focusses on the problems encountered due to the saturation of the photo-multiplier tubes (PMT) in the measurement of high dose values and in the residual signal in the aftermath of these dose assessments.

The dosimetry system implemented at the Individual Monitoring Service (IMS) of Instituto Superior Técnico (IST), Laboratório de Proteção e Segurança Radiológica (LPSR) is based on two Harshaw 6600 readers and on several dosemeter assemblies for the assessment of whole-body, extremity, area and environmental doses. The whole body dosemeter consists of the Harshaw 8814 holder with two-element cards are used for the measurement of the personal dose equivalents Hp(10) and Hp(0.07), based on either LiF:Mg,Ti (TLD-100) or LiF:Mg,Cu,P (TLD-100H).

The TLD system was previously characterized in terms of detection limits, linearity, reproducibility, energy and angle dependences and fading before the implementation into routine work. The linear response of the TLD-100H dosemeters was assessed for dose values between 0.1 and 3 Sv. However, for doses higher than 100 mSv the measured results overestimated (around 20%) the true values and for dose values higher than 1 Sv the system shows a saturation level.

The main reason for this is the quantity of the light emitted by the detectors during the readouts that saturates the PMT circuit. To solve this problem, a filter was inserted to reduce the emitted light by a factor of 10. A new assessment of the linearity was performed until 3 Sv and the results are shown. On the other hand, a study of the residual signal of the dosemeters exposed to high dose values (>500 mSv) was performed and is also presented.

# The Individual Monitoring Service at IST-LPSR

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The laboratory for Dosimetria Individual da Radiação Externa (DIRE) is the Individual Monitoring Service (IMS) of Instituto Superior Técnico (IST), Laboratório de Proteção e Segurança Radiológica (LPSR). The IMS operates since the early 1960s dedicated to the assessment of external radiation exposure. In 1995 a thermoluminescence dosimetry (TLD) system was acquired and introduced for individual and environmental monitoring.

The TLD system is composed of two Harhaw 6600 readers and several types of dosemeters for whole-body (wb), extremity, area and environmental dose assessments. The wb dosemeter consists on the Harshaw 8814 holder with two element cards for the assessment of Hp(10) and Hp(0.07) with sensitive elements of LiF:Mg,Ti (TLD-100) or LiF:Mg,Cu,P (TLD-100H). The extremity dosemeter is the Harshaw EXT RAD for the assessment of Hp(0.07) on fingers with TLD-100H detectors. The area and environment dosemeter is based on the Harshaw 8855 holder for the assessment of H*(10) with TLD-100.

The overall system is periodically calibrated in terms of the different quantities at the Laboratory for Metrology of Ionizing Radiation of IST-LPSR.

The IMS regularly participates in the available Intercomparison (IC) exercises, namely, the ICs organized by EURADOS and other organizations.

At present, the IMS provides wb dosemeters to 3000 workers from 260 facilities (approx.), 70 extremity and 22 area dosemeters to 12 facilities, on a monthly or quarterly period depending on the request. Around 15 sites are quarterly monitored in the country for environmental dose assessment.

In general, the users and facilities are from the medical (80-85%), industrial (10-15%) and research (5%) fields of activity.

The poster presents the main features of the IMS in compliance with the national legislation, the EN ISO/IEC 17025 standard and the results of the participation in the IC exercises.

# Eurados whole body dosimeters intercomparisons: lessons learnt from individual results


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Every two years since 2008, Eurados WG2 has organized a whole body dosemeters intercomparison (IC) in photon and beta fields. This paper presents the information that individual participants can extract from their results. Some particular dosemeters responses are also shown and the possible explanations for such anomalous behaviours are appointed.

The irradiation plan of each IC was designed to allow participants check the performance of some important characteristics of their dosimetry systems:

1. Linearity: dosemeters irradiated to 3 dose values: low, medium and high with the same radiation field;
2. Angular response: irradiations performed in normal incidence and 45º or 60º angles;
3. Energy response: providing irradiations from low energy x-ray, to gamma radiation energy;
4. Response to mixed fields: by irradiating the same dosemeters to 2 different radiation fields;
5. Response to beta radiation: Sr-90 in mixed gamma-beta field was tested in IC2016.

But also very important for the IMS is to show compliance with the standard ISO 14146, and the so called *trumpet curves* that represent the performance limits for personal dosemeters.

In conclusion, all types of dosimeters, regardless of the dosimetry technique, can provide good results if the calibration, reading and dose evaluation procedures are adequate. The analysis of anomalous responses indicates that many systems can reduce the number of outliers by improving their calibration procedures.

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# Simulation of $H_p(0.07)$ beta and photon response of a BeOSL finger ring dosemeter

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During the development of the new BeO based extremity beta detector for application in a BeOSL beta finger ring dosemeter[1], MCNP6 simulations were used for evaluation of different designs without the need of manufacturing prototypes and testing them in the lab. However, ICRU never published beta conversion coefficient from fluence to $H_p(0,07)$ dose. Due to this lack of conversion coefficients it was necessary to independently simulate the $H_p(0,07)$ reference dose in the rod phantom to establish references for various fields used in the simulation. The reference geometry in the rod phantom was based on the ICRP 116 Annex G, with our own assumptions about the properties that were not defined (depth range and shape of the scoring volume). In addition, spectra for BSS2 published by Behrens[2] and spectra for the N-Series as well as for $^{137}$Cs and $^{60}$Co published by Ankerhold[3] were used in the source definitions of the simulations. In the simulations of the BeO based detectors energy deposition in the sensitive material was adjusted for intrinsic OSL efficiency[4]. The simulation tools were first validated with data available for a standard BeOSL photon ring[5].

Design proposals for the beta detector element were transferred from CAD to MCNP to be able to precisely evaluate dosimetric behaviour of each design inside a beta finger ring holder, with the goal to minimise the need for manufacturing prototypes and repeated full characterisation in the lab. Finally, we show comparison of simulations and measurements of the new BeO based beta detector.

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# EURADOS IC2019$_{exteye}$ - A SELF- FUNDED INTERCOMPARISON FOR EXTREMITY AND EYE LENS DOSEMETERS ORGANISED BY EURADOS WORKING GROUP 2

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Aims

The European Radiation Dosimetry Group (EURADOS) has been organising regular, self-funding, external dosimetry intercomparisons (ICs) since 2008 to meet international demand. Participation in these ICs provides the opportunity for Individual Monitoring Services (IMS) to show compliance with their quality management systems, compare results with other participants and develop plans for system improvement. These ICs are entirely self-funded with all irradiation and coordination costs covered by the participation fees. The program, coordinated by EURADOS Working Group 2 (WG2), is intended for dosemeters used routinely for personal monitoring.

Methods

Since 2008 there have now been seven ICs for whole-body dosemeters in photon fields (and one in mixed photon/beta fields), two ICs for extremity dosemeters in photon and beta fields, and two ICs for whole body dosemeters in neutron fields. EURADOS also organises additional ad-hoc ICs, eg for eye-lens and environmental dosemeters. Demand continues to be high with IC2019$_{exteye}$, organised for extremity and eye lens dosemeters, attracting 113 participating systems (68 extremity and 45 eye lens) from 60 institutes with participants from 26 countries around the world.

The scope was for i) extremity dosemeters designed for $H_p(0.07)$ - these dosemeters could be ring, stall or wrist, for wearing on fingers, wrist or ankle, ii) eye lens dosemeters designed for $H_p(3)$ which are worn on the head (not on the chest) and iii) extremity and eye lens dosemeters designed for either photons and betas or photons only. Irradiations were carried out in accredited / primary standard European irradiation facilities in terms of $H_p(0.07)$ or $H_p(3)$ in the ranges: photon energy: 16 to 662 keV, beta mean energy 250 to 1000 keV, dose: 0.5 mSv to 1 Sv, maximum angle of incidence: ± 60º. The paper describes the $IC2019_{exteye}$ set-up, irradiation plan and coordination with a detailed analysis of the results and an overview of the main conclusions.

Results

Out of the total of 113 systems, 57 reported results for photons only and 56 reported both photon and beta. In general, the participants showed a very satisfactory performance with the medians of all $H_p(0.07)$ and $H_p(3)$ response values very close to unity.

Keywords: intercomparison, IMS, extremity, eye lens, EURADOS

# Personal dosimetry in laser generated ultra short pulsed fields: first experience

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ELI Beamlines is one of the pillars of the ELI ERIC (Extreme Light Infrastructure European Research Infrastructure Consortium), the European project aiming at building the next generation of high power lasers for fundamental research and industrial applications.

Several high power lasers are hosted by the ELI Beamlines facility, with varying wavelength or peak power. These lasers are exploited to generate beams of ionizing radiation fields for various research purposes, ranging from primary research of plasma, to X-ray imaging, particle acceleration, or cancer treatment.

Beside ensuring reliable workplace monitoring, it is vital to be able to determine correctly the personal dose received by the personnel. While this task may be straightforward for conventional radiation fields, it represents some challenge for ultra-short pulsed and mixed fields that are generated by lasers. Currently available dosimeters are typically designed for measurements in continuous fields, while the fields generated by lasers are pulsed, with very short pulse length (below 1 ps) and operating in a single shot regime up to the repetition rate of 1 kHz.

The first experiences and lessons learnt from the first months of commissioning of several experimental setups driven by lasers, will be presented and discussed.

# A retrospective analysis of occupational radiation exposure in the medical departments with the highest doses

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## Aims

We analyzed high staff doses in our medical facility to optimize the radiation protection program. During interventional radiology procedures, staff doses can be high. Also in a nuclear medicine department, staff is exposed during preparation and injection of radiopharmaceuticals and while preparing the patient for imaging. To gain more insight in trends and differences in occupational exposure, doses were anonymously coupled to worker information such as profession, years of experience and work regime.

## Materials and methods

Doses were collected using whole body Hp(10) personal dosimeters (TLD type Harshaw 0110/8814). During interventional radiology procedures dosimeters were worn above and under the lead apron. All dosimeters were read out in the accredited dosimetry service of UZ Leuven using a Harshaw 6600 Plus TLD reader. The monthly doses collected for the years 2013 up to 2018 were anonymously coupled to information about profession, years of experience and work regime. The study was approved by the Research Ethics Committee UZ/KULeuven.

## Results and discussion

For the interventional procedures we found a downward trend in the average annual dose resulting in 1,4 mSv (range 0-6,3 mSv) for physicians and 0,5 mSv (range 0-2,6 mSv) for nurses in 2018. To eliminate a possible influence of work regime a new analysis was done, based only on doses from full time workers (n=27). This lead to a decrease in the average yearly dose of 3,5 mSv in 2013 to 1,6 mSv in 2018 for physicians and 1,4 mSv in 2013 to 0,6 mSv in 2018 for nurses. The data were further analysed for work experience (<5y, 5y-10y, >10y), showing reduced doses for more experienced nurses.

In nuclear medicine the nurses injecting the radiopharmaceuticals and aiding the patient during the imaging procedure had the highest annual doses. For a work experience of <1year (n=5) the average annual dose was 3,6 mSv (range 0,5-6,3 mSv) compared to 1,9 mSv (range 0,1-5,1 mSv) for the technologists with >1year working experience (n=11).

All doses were well below the dose limit. However, the yearly dose constraint of 6 mSv was reached for a few interventional radiologists with a high workload and also for nuclear medicine nurses injecting predominantly PET-tracers, or having less work experience (<1y).

## Conclusion

Since workload did not decrease and reported values were normalized for a 100% working regime during the 6 years, the downward trend in staff doses in the cathlab and nuclear medicine may be explained by work experience and training in the use of the appropriate radiation protection measures available in the workplace. Statistical analysis of the dose distributions can confirm the observed trends, but would be ideally done on larger datasets by collaborating with other dosimetry services of medical facilities.

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## Preliminary Investigation of the Deuterium-Tritium Neutron Generator Shielding Design at NSIL

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A Deuterium-Tritium (DT) neutron generator producing 14 MeV fast neutrons, needs sufficient shielding to limit dose rates to the stipulated values both inside and outside of the facility [1], [2]. Neutron facilities with such fast neutron generators are always concerned about shielding design, cost, weight and of course, dose limitation[3]. The optimal shielding design for DT will differ depending on the specific facility, its room arrangements and equipment, and on the budget for the project. In order to guarantee the compliance with the safety rules for public and workers, a detailed knowledge of the radiation dose rates around the neutron generators is essential.

In this study, simple radiation transport models for the DT generator in MCNP6.2 and PHITS codes have been validated. Further, several conceptual shielding design options have been assessed as candidates for the DT shield in the Neutron Science Facility of NSIL. The study considered various shielding materials and geometries and aimed to minimize shielding costs and weight while fulfilling the dose-rate criteria. Although, none of the design candidates have been realized, the analysis contributes to understanding of the shielding design options and the radiation doses investigation at workplaces for sealed tube neutron generators.

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# Development of *in vitro* Gamma Spectrometry as a complementary technique for *in vivo* measurements in routine and emergency. Application to a real case of contamination with $^{177}Lu$

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CIEMAT Bioelimination Laboratory is authorized by the Nuclear Safety Council and accredited by ENAC (ISO17025:2017) to quantify the activity of alpha and beta emitters in excreta samples (urine and faeces) from workers exposed with risk of internal contamination. This work focuses on the development of the gamma spectrometry technique for the determination of gamma emitters in urine samples in routine and emergency scenarios. This technique can be used to complement *in vivo* measurements carried out in CIEMAT Whole Body Counter (WBC), the authorised and accredited method to quantify gamma emitters.

A p-type coaxial high purity Germanium semiconductor (HPGe) detector with an efficiency > 30%, resolution < 2 keV and an energy range from 50 keV to 10 MeV was employed. Energy and efficiency calibration protocols have been established studying different measurement geometries: Nalgene, Duchess and Marinelli. Several parameters have been checked (geometries, sample volume, counting time, background, efficiencies, detection limits…) in order to establish the optimal measurement conditions. Validations and quality control activities have also been carried out.

In order to validate the methodology, Bioelimination Laboratory has participated in an international intercomparison exercise, analysing three urine samples: 2 spiked samples and one non-spiked sample. $^{40}K$ was measured and quantified in the three samples. Laboratory results fulfilled the ISO 28218:2010 performance criteria, and very good results were obtained for $^{54}Mn$, $^{60}Co$ and $^{40}K$ isotopes with bias from -3.67 to -8.24%. These results guarantee the validation and the accuracy of this method.

Finally, the developed methodology was applied to a real contamination scenario from a worker involved in an incident with $^{177}Lu$. Two spot urine samples were measured and analysed by gamma spectrometry. These bioassay results support the intake evaluation based on *in-vivo* (total body) monitoring data obtained at CIEMAT WBC.

This work is part of the Master's Dissertation carried out by M. Barrio in CIEMAT Bioelimination Laboratory.

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# ANOMALY DETECTION OF TLD GLOW CURVES USING THREE DIFFERENT MACHINE LEARNING CLASSIFIERS

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A novel method for the detection of anomalous glow curves (GCs) is presented. The method uses three machine learning (ML) classifiers in order to detect abnormal GCs shapes. A high detection performance rate for either 'anomalous' or 'regular' GC shape class of almost 98% is achieved, depending on the classifier type.

Anomalies in GC shapes, if ignored, might significantly change dose estimation. Moreover, many external dosimetry laboratories (EDLs) maintain quite busy lines of TLD readings, so they will benefit from a fully automatic process that that will automatically detect anomalous GCs.

The method in this research was developed in the EDL of the Radiation Safety Department at the Soreq Nuclear Research Center. We compared results of three variants of ML classifiers: Artificial Neural Networks (ANNs), Support Vector Machines (SVMs), and weighted SVMs, all of which have been extensively employed in ML classification applications in numerous fields of research.

The application of the above three ML classifiers includes the following three stages:

- Data tagging, in which each input in a data set is manually labelled by a laboratory technician with an appropriate class label, in our case either 'anomalous' or 'regular'.
- Training method which trains the algorithm over this set of tagged input in order to obtain an optimal set of algorithm parameters.
- Predicting the classification of some new data using the trained classifier.

In order to test the developed classifiers, we ran each of them separately on 3654 untagged GCs, and let them classify the GCs into either ‘regular’ or ‘anomalous’ GCs. Results showed that all three methods produce similar results in terms of detection accuracy – between 96.3% to 97.7%. We chose to adopt the weighted SVM classifier since it had the lowest false negative rate among the three methods, which means it had the fewest cases that the classifier classified a GC as 'regular' one while it fact it was an "anomalous" one.

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# TLDetect - A NOVEL SOFTWARE FOR DETECTING AND CORRECTING ANOMALOUS LiF:Mg,Ti GLOW CURVES FOR EXTERNAL DOSIMETRY LABORATORIES

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As a part of improving the External Dosimetry Laboratory (EDL) processes at Soreq Nuclear Research Center (SNRC), we characterized a new algorithm and built a corresponding software tool - TLDetect, which enables automatic detection of anomalous Glow Curves (GCs) and later for the EDL staff to better identify and treat the anomalous GCs. The automatic detection of anomalous GCs is an extremely important task for EDLs due to the following reasons:

- Anomalous GC might introduce dose over-estimation into the final dose report
- In high-volume EDLs, staff need to examine tens of thousands of GCs a month
- Being able to automatically spot anomalous GCs can improve the quality assurance of the EDL by enhancing the detection repeatability and its objectivity, without any dependence on the EDL staff examining the curves

TLDetect uses both Machine Learning (ML) tools and deterministic methods to identify the anomalous GCs. It first applies on the GCs data a support vector machines classifier, which is a ML algorithm that can classify the data into two different classes – ‘regular GC’ or ‘anomalous GC’.
After filtering out all the ‘regular’ GCs, several conditions are being checked in order to try to classify the rest of ‘anomalous’ GCs into four different anomalous classes:

i. GCs with spikes
ii. GCs with extraordinary intensity glow peak at either low or high temperature region.
iii. GCs that have either too narrow or too wide glow peaks.
iv. GCs with glow peaks, which are shifted either to the low or to the high temperatures region.

After classifying the GCs into the above 4 classes, the algorithm tries to automatically amend the GCs which can be fixed. It asks the EDL user whether to automatically fix the GC or to allow a user intervention. Moreover, the algorithm saves the statistics of all GCs in terms of dosimeter IDs, classes and sub-classes that the dosimeters were classified into, time stamps, and some more internal statistical data of the algorithm itself, for later analysis of trends in system performance. Certain data statistics can be of great help to spot some hardware problems in the TLD readers, which may be closely related to specific GC shape anomalies. On top of that, the new tool will save a lot of precious time to the EDL staff by automatically spotting the anomalous GCs out of all the GCs, without the need to manually review all the GCs.

# LNMRI/IRD's TNF2 inverse square law for a virtual source

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For a point source of radiation, emitting isotropically in all directions, the radiation fraction that reaches the detector will depend on the distance between it and the source, as well as the detector sensitive area facing it. By the relations that geometry provides, it can be demonstrated that the radiation intensity received by the detector varies inversely with the square of the source-detector distance, being known as the "Inverse Square Law" (ISL).
Although the existence of a point radiation source is necessary for this law to be valid, in some cases where there is more than one source, it is possible to use this law as long as a point is determined where a virtual source can be positioned to forward points corresponding to the ISL.
In the LNMRI/IRD's TNF2 thermal neutron system, which has 4 Am-Be sources, as there is a neutron flux in a direction perpendicular to the front face of the cube, it is possible to adjust a curve and determine both the position of the virtual source and the points that comply with the ISL.
To use the system for testing neutron monitors, a curve with the points corresponding to the ISL was plotted. For this purpose, ambient dose equivalent rate values were obtained by simulation at several points along the central axis of the frontal face and a function corresponding to the distance inverse square and position of the virtual point source "X" was determined corresponding to ISL.
The values were derived to ambient dose equivalent rate, along the central axis. Commercial monitors NRD-600 and WENDI and the MCNPX simulation at the same points were used. The values in each of the measurement series were compared and plotted in graphs, showing the MCNPX curve is in good agreements to other two experimental curves, validating results and allowing system to be incorporated into the TNF2 and used in calibrations.

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# CD FLAT RING DOSEMETER FOR EXTREMITIES AND EYE LENS DOSIMETRY

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The growth of radiopharmacy and interventional radiology has created the need to increase the control of extremities and the eye lens, with specific dosimetry. In this context, centro de dosimetría, s.l. has designed, manufactured, and patented a versatile dosemeter to control extremities and eye lens with the following functional properties:
●Single use to improve hygiene and break down any radioactive contamination. ●Hermetic ●Extensive identification of user, period of use and department ●Ergonomic, flexible fit ● Alternative options for positioning the eye lens dosemeter (headband, cap, glasses) ● Ability to discriminate beta radiation from RX-Gamma.

We also highlight the following technical aspects: ●The ring capsule has the same polypropylene filtration as the one we used previously (Radpro) which we had widely tested(1,2,3,4). ● Two detectors can be housed one above the other in the ring capsule. ● The flat shape and the broad surface provide easy laser marking. ● The capsule is ultrasonically welded to ensure hermeticity. ●The same mould can provide a bulge for joining a clamp as additional support for the eye lens dosemeter. ●The dosemeter has been very well accepted by users.
Both configurations have been verified at the IC2019exteye intercomparison and recently in 2021, additional calibrations have been made to verify the radiation performance requirements of the current IEC(6).

RESULTS & CONCLUSIONS
The dosemeter is considered operational for both measurements since the results fulfill the trumpet curves(5) criterion for both configurations. The most unfavourable result for the eye lens dosemeter was $^{137}$Cs, and for rings W80 and $^{90}$Sr angular irradiation.
Application of the calibration factor provided by the ratio of the readings of the two pellets, gives a dose deviation under 15% for the full energy range except for a few cases when the reading ratio does not allow identification of specific energy. In those cases, an additional algorithm is applied, and for relevant doses checking with the customer is recommended.

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# Uncertainties associated with direct thyroid measurements using an NaI(Tl) survey meter: the effect of anatomical differences among individuals

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Direct thyroid measurements to determine the radioiodine content in the thyroid could be required for assessment of early internal doses in affected populations after a major nuclear accident. Detectors used for these measurements are normally calibrated using a physical phantom that mimics the human neck including the thyroid; however, there is uncertainty associated with the measurements due to differences between the phantom and actual subjects[1)], also to anatomical differences among individuals[2)]. The present study recruited 24 healthy Japanese volunteers (15 males and 9 females) aged 20-60 years old and performed MRI scanning on their neck parts. Personalized computational phantoms were created by voxelizing each slice of the T2-weighted MR images into the airway, the soft tissue, the thyroid and the surrounding air. As a result, the thyroid volume was determined to be 14.9 ± 4.0 $cm^3$ for male volunteers and 13.8 ± 4.0 $cm^3$ for females, results that were confirmed to be reasonably consistent with those estimated from their ultrasound images. We implemented radiation transport simulations of direct thyroid measurements in conjunction with these phantoms in order to evaluate the conversion factor from the dose rate of an NaI(Tl) survey meter (TCS-171/172, Hitachi Aloka Medical, Ltd.), which is widespread in Japan, to $^{131}I$ in the thyroid. Conversion factors for the male and female volunteers were 27.6 ± 6.4 kBq/(µSv/h) and 25.7 ± 5.8 kBq/(µSv/h), respectively, when the probe of the survey meter was placed on the anterior neck. These values were found to be smaller than those obtained for the Japanese male (JM) and female voxel phantoms (JF)[3, 4)], 35.2 kBq/(µSv/h) and 28.8 kBq/(µSv/h), respectively; one of the reasons is that thyroid volumes of JM and JF (i.e., 20.8 $cm^3$ and 16.0 $cm^3$, respectively) are generally larger than those of the volunteers. These results indicate that conversion factors based on JM and JF would offer moderate overestimates of the $^{131}I$ thyroid content for Japanese adult subjects and that the influence of the thyroid volume on dose assessment should be considered in determining the organ-specific individual risk.

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# Response of current dosemeters to new operational quantities

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The ICRU proposed new operational quantities for external radiation exposure, which are defined in close relation to the protection quantities[1]. The new quantities are named personal dose, $H_p$, personal absorbed dose to the lens of the eye, $D_{p\ lens}$, personal absorbed dose in local skin, $D_{p\ local\ skin}$, which are related to personal monitoring, and ambient dose, H*, directional absorbed dose to the lens of the eye, $D'_{lens}$, and directional absorbed dose in local skin, $D'_{local\ skin}$, for area monitoring. These will replace the current operational quantities, i.e. personal dose equivalents, $H_p(10)$, $H_p(3)$, $H_p(0.07)$, for personal monitoring, and ambient dose equivalent, H*(10), and directional dose equivalents, H'(3), H'(0.07), for area monitoring, which are defined to estimate the protection quantities more conservatively. Because of differences between the new and old operational quantities, the currently used dosemeters may not be ideal for measurements of the new quantities. The differences depend on the type, energy and the direction of the incidence of particles, and also on the specific quantity, related to the deep or shallow dose. To investigate the relative responses of the existing personal and area dosemeters with respect to the measurement of the new quantities, various irradiations using photon and neutron spectra were performed in laboratory conditions. No eye dosemeter was included into tests, but several active and passive photon dosemeters designed to $H_p(10)$, H*(10), $H_p(0.07)$ and H'(0.07) measurements and also active neutron dosemeters for $H_p(10)$ and H*(10) were tested. The results showed that introduction of the new quantities into practice will require some changes. A recalibration will be essential for $H_p$ and H* measurements of photons with the energy higher than 50 keV. For lower energies, a redesign of calculation algorithm or construction of dosemeters will be necessary. For $D_{p\ local\ skin}$ measurements, current dosemeters can be used without modifications, but for purposes of $D'_{local\ skin}$ measurements, correction of calculation algorithm can be required. Some neutron dosemeters overestimated for thermal neutrons, this effect increased after implementing $H_p$. However, in general it can be concluded that no fundamental changes will be needed in the case of neutron dosemeters.

The work was supported by the Ministry of the Interior of the Czech Republic within the frame of the project VI20192022156 "Dosimetry for radiation accidents and incidents in the context of new operational quantities for external radiation".

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# Biokinetic model analysis with DTPA administration for the case of accidental inhalation of actinides in Japan

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Accidental inhalation of the actinides (e.g., $^{239}$Pu and $^{241}$Am) occurred among five workers at the Japan Atomic Energy Agency in 2017[1)] and Ca- and/or Zn-DTPA (Diethylene Triamine Pentaacetic Acid), the decorporation agents for the actinides, were intravenously administered to the workers. The provisional effective dose of the worker who was considered to have the greatest exposure was conservatively evaluated to be 100-200 mSv, based on the results of both lung monitoring performed on the third day after inhalation and the bioassay of faecal samples collected during the first five days after inhalation. Chelation therapy with intermittent DTPA administration continued for this worker for more than one year, and bioassays of urinary samples collected before and after each DTPA administration were performed. In this study, to obtain a reasonable understanding of the behaviour of the actinides incorporated into the subject's body for sophisticated internal dose assessments, a biokinetic model for actinides with DTPA administration was constructed according to the European Coordinated Network on Radiation Dosimetry (CONRAD) approach[2)], which simulates the process by which Pu/Am-DTPA is produced by reaction between Pu/Am and DTPA in the same compartment. The Occupational Intake of Radionuclides (OIR) series[3)] published by International Commission on Radiological Protection (ICRP) was used for the Pu/Am model, and the reaction with DTPA was assumed to be caused in blood, interstitial fluids in the soft tissues and lymph. Intake conditions such as the chemical form and particle size were determined from the ratio of the $^{241}$Am content in the lungs to that in the urinary and faecal samples, respectively. As the urinary excretion of $^{241}$Am is largely affected by DTPA administration, the $^{241}$Am content in the urinary sample obtained before the first DTPA administration was used to determine the chemical form. Here, we present a comparison of the time variation of the $^{241}$Am urinary excretion between the measured values obtained from the bioassay and the model prediction obtained by optimizing values of the parameters including the reaction rate constant for each compartment where $^{241}$Am-DTPA is produced.

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# Establishment of Hp(3) Calibration System for Eye Dose Monitoring in Taiwan

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In response to the ICRP's amending the occupational exposure limit for the lens of the eye, it is necessary to develop accurate dose measurement equipment of the eye lens in Taiwan and provide appropriate eye dose monitoring. For this purpose, we established the Hp(3) dosimetry calibration system for the lens of eyes through domestic dose traceability system to accurately assess the dose received by the eye lens.
Therefore, the National Radiation Standard Laboratory of Institute of Nuclear Energy Research (NRSL/INER) established the X-ray beam quality using the self-made free air chamber referring to ISO 4037. And we determined the Hp(3) occupational quality radiation field, measured monitoring factors of ionization chamber used to converse real-time reading and dose, and evaluated the uncertainty of the calibration system to establish the eye lens dose calibration system. After the dose calibration system was established, DOSIRIS headband typed dosimeters were studied for the characteristics of the eye lens dosimeter including dose linearity, energy- and angular- dependence. The results could be applied in performance testing of the next stage. The new calibration system will be implemented in 2022.

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# A new direct-reading neutron personal dosemeter

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**Abstract text:**

The use of active personal dosemeters (APD) in physical surveillance of radiation workers is likely to increase, as they can effectively operate as ALARA or alarm devices, providing immediate feedback to the worker.

As far as the neutron field is concerned, the development and dissemination of active personal dosemeters is still experiencing serious barriers, mainly connected to the difficulty in providing accurate dose response over ten or more orders of magnitudes in energy.

Within the DOIN project (Dosimetro indossabile per neutroni), INFN-Italy patented a design of a neutron APD (APD-n).

The DOIN device, designed relying on FLUKA code, provides ”rough" spectrometric information on the field. This enables to offer an improved dose response over a wide range of workplaces.

This communication describes the performance of DOIN APD-n in view of possible industrial applications.

# Development of ring radiation dosimeters using 3D printing

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Extremity radiation monitoring using ring dosimeters is an important tool for the assessment of occupational exposures to staff at workplaces in interventional radiology (IR) and nuclear medicine. At the Radiation Monitoring Laboratory of the Federal University of Pernambuco, additive manufacturing technologies (3D printing) using high resolution stereolithography (SLA) have been extensively tested for radiation monitoring applications. This approach is particularly interesting not only due the near equivalence to water of the resins, but also due to the sub-millimetric reproducibility, easiness for production adaptiveness after user feedback, integration to existing protective equipment, and the possibility of manufacturing flexible, wearable and sterilizable dosimeter holders, and therefore improving the adherence of medical staff to use the dosimeters, which still hinders the widespread use of monitoring equipment. The aim of this paper is to characterize the response of a ring radiation monitor developed using high-resolution 3D printing holding a BeO optically stimulated luminescence (OSL) detector element. The energy and angle dependence responses were evaluated and compared with a commercial ring monitor.

Two different resins, hard and flexible, are used on the casing to achieve both radiation and mechanical appropriateness of the ring monitor. The ezClip BeO OSL dosimeter from Dosimetrics GmHb is the sensitive element. Comparisons were made with the commercial ring casing with the same type of BeO detector. The OSL reader model is EC compliant and PTB type tested (Physikalisch-Technische Bundesanstalt, Braunschweig, Germany).

Energy and angular dependence were performed using ISO Narrow series radiation beams with energies from 24 keV to 164 keV and S-Cs137 (662 keV). Tests by illumination of exposed rings over a week confirmed bleaching of the OSL signal from ambient light is negligeable.

Results showed that the printed ring dosimeters easily comply to IEC 62387 limits for energy and angular. Energy dependence at 0° is in within -15% to +15% in the range of 24 keV to 662 keV and a slightly lower energy dependence between 1.8% up to 4.6% was achieved when compared to the commercial ring dosimeters. At 33 keV and 60° the maximum energy response of the printed dosimeter was +34% compared to +31% of the commercial ring dosimeter. This was considered acceptable to within other experimental uncertainties and is due to the design of the closing mechanism.

This work showed that high resolution stereolithography can be used for tailoring novel and highly adaptive ring radiation monitors. Coupled to BeO OSL dosimeters, a new ring dosimeter was made that complies to IEC standards.

# Dosimetric methods at the proton irradiation facility at the AIC-144 cyclotron at IFJ PAN in Krakow

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At the Institute of Nuclear Physics of the Polish Academy of Sciences in Krakow (IFJ PAN), proton irradiation station for tests the electronics radiation hardness has been prepared. The irradiation station allows deliver the proton beam with an intensity from 5e6 proton / cm2 * s to 1e10 proton / cm2 * s. Adjustable, collimated radiation fields from 1 cm to 12 cm in diameter for a passively scattered beam and a field of 25 cm x 35 cm has been predefined. The dose rate depends on the irradiation station configuration and ranges from 1 mGy / s to 17 Gy / s.

Such defined ranges of radiation field and flux of proton beam force the use of dosimetric methods that allow both to verify the parameters of the radiation field and to perform a dose calibration of the proton irradiation stations (proton beam fluxes and fluence determination). The aim of the work is to present the dosimetry methods, to provide the ranges in which particular method can be used and to show the consistency of the results between various dosimetry methods.

A group of active and passive dosimetric methods has been prepared. Active methods are based on measurements with ionization chambers connected to reference-class electrometers. These methods enable the facility calibration and beam monitoring in the full range of intensities.
Gafchromic EBT3 films and 2D thermoluminescent foils were used to determination beam homogeneity, for radiation field size verification, and for lateral dose distribution determination. Active systems based on plastic scintillators and a CCD camera (ProBimS, LynX) are also available for the verification of the radiated field and the setting of the beam position.

On-line dose monitoring is performed by ionization chambers connected to the reference class electrometer calibrated in accordance with the recommendations of the IAEA TRS-398 Code of Practice [1]. To verify the dose delivered during the sample irradiation, dosimetry methods based on TLD detectors (in the range from 1 mGy to 5 kGy) and the alanine (in the range from 1Gy to 120kGy) were used.

A set of ready-to-use dosimetric methods for electronics radiation hardness tests allow to determine the delivered dose with an accuracy of 5%.

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# Preliminary characteristics of neutron field from the Song150 neutron generator for calibration of radiation protection devices – neutron and gamma doses

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Dosimetric laboratories involved in the calibration of dosimetric equipment need good standard sources to create their reference fields. From the point of view of regulatory bodies, measurements with devices calibrated in such a field are crucial for the evaluation of radiation risk generated by high-energy photon or electron beams generated in industrial/medical linear accelerators [1].

The main aim of this work was to investigate the feasibility of the facility with the D-T type neutron generator for calibration of commercially available neutron area monitors in both Anderson-Braun and Lake types of construction [2].

To extend our calibration capabilities In National Centre for Nuclear Research [3] we attempt to characterize a radiation field around the Song150 neutron generator with the nominal neutron emission rate in the order of $10^8$ n/s and energy 14 MeV. Experimental measurements of neutron fluence and its energy distribution were performed with a set of activation foils. Absorbed doses of gamma and neutron radiation were measured with REM-2 and GW2 ionization chambers [4] used as the set of twin detectors. Furthermore, from additional REM-2 measurements running in recombination mode we were able to determine recombination quality factor, $Q_4$, being the estimation of radiation quality factor, Q. The results were compared to ambient dose equivalent, H*(10) values, obtained from ICRU conversion coefficients.

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# The Revision of ISO 8529 - Neutron reference radiation fields

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**Abstract text:**

ISO International Standard 8529, "Neutron reference radiation fields" provides basic guidance on neutron metrology for both primary and secondary calibration laboratories. Its three parts, namely Part 1: "Characteristics and methods of production" [1], Part 2: "Calibration fundamentals of radiation protection devices related to the basic quantities characterizing the radiation field" [2] and Part 3: "Calibration of area and personal dosimeters and determination of response as a function of energy and angle of incidence" [3], are under revision by ISO/TC 85 / SC 2 (Radiological Protection) / WG 2 (Reference Radiation Fields) / SG3 (Neutrons).

The revised Part 1 was published in November 2021 as a Second Edition [4]. The changes introduced in the new version of the standard will be discussed. Revised spectra, which take into account new data, and which have implications for fluence to dose equivalent conversion coefficients, are the main changes.

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# Spectrometry Study from $^{252}$Cf Moderated in Water Neutron Source

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The Neutron Metrology Laboratory (LN) is reference in Neutron Metrology, being responsible for custody and maintenance of the Brazilian Neutron Fluence Standard, as well as the quantification of neutron fluence. Among the ISO8529 reference fields, LN maintains and disseminates reference values for $^{241}$AmBe, $^{241}$AmB, $^{252}$Cf and $^{252}$Cf + D2O sources. The characterization and availability of new fields strengthen performance type tests of various devices and irradiations of various samples. The purpose of this study was to determine the spectrum range, values of fluence, ambient and individual dose equivalents coefficients produced by a source of $^{252}$Cf positioned in the centre of a steel spherical cavity filled with water. Fluence values were determined using the bonner multisphere spectrometer for 84 neutron energies ranging from $10^{-9}$ MeV until 20MeV. The values of personal dose equivalent and ambient dose equivalent were also determined, and their uncertainties were estimated.

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# Methodology to obtain quality factors for neutrons beyond 19.5 MeV using standard Monte Carlo codes.

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For evaluating the radiobiological damage of neutrons, the quality factor Q as a function of the energy must be applied to the energy distribution of the secondary charged particles, which appear from the interactions between neutrons and matter. This factor is indirectly used when conversion coefficients from a physical quantity-to-operational quantity for neutrons are applied. Siebert and Schumacher 1995 work [2] is usually taken as reference for neutron quality factors for energies up to 19.5 MeV, but there are several works where dose equivalent conversion coefficients for neutrons are provided for higher energies, for instance [2, 3, 4]. The recent International Committee for Radiological Units (ICRU) report 95 [5] presents new values of the conversion coefficients from fluence to several quantities for many incident particles, but no values for Q are reported.

At present, there are situations (like out-of-field patient doses in proton therapy) where the energy distribution of the neutron fluence extends beyond 19.5 MeV and where the fluence-to-ambient dose equivalent conversion coefficients for neutrons are not usable. In this case, the quantity neutron dose equivalent must be obtained from the neutron fluence energy distribution via the neutron quality factors and fluence-to-kerma conversion coefficients for neutrons [6], but there is a lack of reference data for Q at these energies.

In this work we intend to, first, review the neutron quality factor and to propose a standard methodology which can be applied to any Monte Carlo code to compute it and, second, to calculate with this methodology, using both MCNP6.2 and PHITS, the quality factors for neutrons up to 5 GeV.

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# CHARACTERIZATION OF $^{238}$PuBe SOURCES USING BONNER SPHERE SPECTROMETER

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The system for the neutron metrology is a procedure that requires the use of specific methods along with different types of detector systems so that the desired values are finally obtained. The methods used in the laboratory of Neutron Metrology (LN) are the manganese sulphate bath (BSM), the system of the Precision Long Counter (PLC) and the Bonner Multisphere spectrometer. This experiment aims to perform technical measurements by experimental means of neutron-emitting radioactive sources to characterize the neutron spectra. Bonner Multisphere Spectrometer with europium-enriched lithium iodide scintillating detectors [$^{6}$LiI (Eu)] and $^{238}$PuBe sources with activities of 296, 444, 518, 1850 GBq (8, 12, 14 and 50 Ci, respectively) was used. Each source used in the low scattering laboratory was submitted to a total of 21 measurements among the different diameters of the spheres of the EB system, thus enabling the capture of different energetic bands, counts, time and area of Counting among other factors and quantities referring to each radioisotope source. All these factors together with specific equipment and computational programs, such as NEURALN[1][2], assist in the deconvolution or unfolding of a neutronic spectrum of fluence x energy for each radioactive source.

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# Interactive software tools for personalized dosimetry simulations

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Computer simulations have become a primary driver of dosimetry and radiation protection research. Thanks to more detailed computational phantoms geometries, improved interaction models and faster computers, our simulation capabilities have increased exponentially both qualitatively and quantitatively, up to a point where it is nowadays possible to achieve truly personalized dosimetry simulations. However, these improvements have also led to a steep growth of complexity, making it harder for researchers not only to generate simulation input files, but also to read-out information throughout output files and interpret simulations results.

For this reason, at SCK CEN we developed software tools for easing the creation of personalized dosimetry simulations and for automatizing the read-out and analysis of simulation results. With this work, we present two software tools: the Interactive Posture Program (IPP) and the Interactive Source Simulator (ISS). IPP allows to interactively change the posture and tune the anatomical features of flexible computational phantoms of the Realistic Anthropomorphic Flexible (RAF) family, and it can generate, run, and analyze simulations with particle transport codes such as MCNP, Geant4 and GATE. The ISS software allows to easily generate thousands of simulations for comprehensive multi-parametric studies, by automatically varying the definition of radioactive sources, such as location, dimension, emission spectrum, direction of emission, filtration and more. Besides generating simulation input files, our software tools can also automatize the read-out and interpretation of simulation results, effectively aiding researchers and enhancing our capabilities of generating personalized simulations.

# Results of extremity dosimetry based on the measurements at Laboratory of Individual and Environmental Dosimetry at IFJ PAN

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Laboratory of Individual and Environmental Dosimetry (LADIS) at the Institute of Nuclear Physics (IFJ PAN) applies MTS-N (LiF:Mg,Ti) thermoluminescence (TL) detectors for individual monitoring in terms of $H_p(0.07)$ in photon radiation fields. Standard TL detector, 4.5 mm diameter and 0.7 mm thickness, is placed in the bar coded plastic ring holder under the 0.1 mm thick plastic cover. The ring holder can be adjusted for any finger size. TL detectors are readout in an automatic RADOS readers (Mirion Technologies (RADOS) Oy) [1].
The dosemeter's response to various routine sterilisation methods routinely used have been tested with positive results [2].
Extremity ring dosimetry measurements are performed at LADIS laboratory for more than twenty years, with accredited procedure since 2002. According to the quality system based on PN-EN ISO/IEC 17025:2018-02 standard, $H_p(0.07)$ personal dose equivalent is measured over the range of doses from 0.1 mSv to 1 Sv.
Years of experience allow the laboratory to analyse the levels of doses received by workers in medical and industrial institutions cooperating with LADIS laboratory. Yearly, more than 35 thousand of extremity measurements have been performed for about 1000 institutions in Poland on the quarterly basis.
According to the internal classification, the radiation workers under control have been divided into few groups. Our analysis indicated that most of the doses received by exposed workers were on the level of the natural radiation background, but some of them exceeded the dose limit. The results showed that about 60% of $H_p(0.07)$ doses were below 0.1 mSv/quarter, however some very high doses were also registered. The highest dose level above 100 mSv was observed in nuclear medicine.
The percentage of overdoses was relatively small but looking at individual cases the risk of irradiation was noticeable. Therefore constant monitoring is always necessary.

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# Past, presents and future of thermoluminescent dosimetry in Poland – a review article

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The beginnings of thermoluminescence dosimetry date back to Maria Skłodowska-Curie observing the glow of natural calcium fluoride during heating and early exposure to ionizing radiation from a radium source [1]. The 1950s, with Farrington Daniels, saw the first application for dose measurement and dose levels from various types of ionizing radiation [2]. In the 1960s, the development center of thermoluminescence dosimetry in Poland was the Dosimetry Laboratory of the Institute of Nuclear Physics leaded by Tadeusz Niewiadomski [3]. In the following years, the original LiF: Mg, Ti (MTS-N) detector was developed, and then the highly sensitive LiF: Mg, Cu, P (MCP-N). Currently, all dosimetry services in Poland use both types of detectors on a mass scale in individual, environmental and clinical dosimetry. The paper will also present the non-standard applications of thermoluminescence in determining the elapsed exposure time, second readout of doses, planar 2D readouts and the development of on the eye lens dosimetry. The presented work will be a review, it will contain the achievements of the 50-year history of thermoluminescence dosimetry in Poland as well as the present and future development elements.

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# Occupational exposure to ionizing radiation in Poland based on dose measurements at Laboratory of Individual and Environmental Dosimetry

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Accredited Laboratory of Individual and Environmental Dosimetry (LADIS) at IFJ PAN provides dosimetric service for individual dosimetry in terms of personal dose equivalent $H_p(0.07)$ and $H_p(3)$ for photon fields and $H_p(10)$ for photon and neutron fields over the range from 0.1 mSv to 1 Sv, air kerma $K_a$ over the range from 30 μGy to 1 Gy and environmental dosimetry in terms of ambient dose equivalent $H^*(10)$ over the range from 30 μSv to 1 Sv.
Currently, LADIS as the biggest dosimetry service in Poland performs over 50000 measurements yearly for individual radiation workers and environmental measurements at working areas or in the natural environment locations, for over 10800 institution.
In all measurements thermoluminescence detectors (TLD) developed at IFJ PAN are applied: the well-known standard MTS-N (LiF: Mg, Ti) for $H_p(10)$ and $H_p(0.07)$, and the high-sensitivity MCP-N (LiF: Mg, Cu, P) for $H_p(3)$, $K_a$ and $H^*(10)$. All the dosemeters/TL detectors are readout in an automatic RADOS readers (Mirion Technologies (RADOS) Oy) [1].
The dose measurements using whole body, extremity and environmental dosemeters started in 2002 at LADIS are an excellent basis for observing trends in radiation protection and for checking the changes in the dose levels in a various ranges. The results from eye-lens dosemeters applied to routine measurements as the first dosimetry service in Europe in 2012 are very valuable [2].
Most of the dose measurement results are at background level, but doses above the dose limit have also been reported. The results showed that more than 80% of $H_p(10)$, 60% of $H_p(0.07)$ and 30% of $H_p(10)$ doses were below 0.1 mSv/quarter, however some very high doses were also registered. From these results, it can be concluded where the greatest radiological exposure can occur and try to identify the main activities leading to the highest radiation exposures in Poland.
LADIS laboratory regularly participates with positive results in the international intercomparison exercises organized by the European Dosimetry Group EURADOS in order to validate the performance of dosimeters used in the Laboratory, like in the last one for whole body dosimeters in 2020.

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# THE HIGHEST DOSES FROM IONIZING RADIATION RECORDERED IN THE LABORATORY OF INDIVIDUAL AND ENVIRONMENT DOSIMERTY–ANALYSIS OF THE RESULTS FROM WHOLE BODY AND FINGER RINGS DOSEMETERS

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Laboratory of Individual and Environmental Dosimetry (Polish acronym LADIS) is nowadays the largest dosimetry service in Poland providing the broad spectrum of measurements of individual doses from ionizing radiation. The dosimetry is based on thermoluminescent detectors MTS-N (LiF;Mg,Ti) and MCP-N (LiF;Mg,Cu,P), developed at the Institute of Nuclear Physics in Krakow in 60's and 80's [1]. In 2002 laboratory obtained accreditation certificate according to the EN-PN- ISO/IEC 17025 standard from Polish Center for Accreditation. Laboratory provides services for almost 11000 various institutions. The largest occupational group are radiologist (almost 60% measured doses) the other numerous cases are industry and oncology centers workers. It should be noted that employees of nuclear medicine departaments take about 2% of the examined cases, however, the permissible doses are often exceeded in this group. Quarterly measurements were carried out and used to estimate doses. Moreover, the laboratory performs quality assurance tests of X-ray machines for medical purposes.

The paper presents results collected by LADIS in the years 2014-2020. In nuclear medicine and interventional procedures, the staff might be exposed to significant radiation doses, particularly to the extremities. The most common cases of exceeding the dose for the whole body are in industry. The monitoring data were calculated using fully automatic hot-gas TL RE-2000 readers produced by Mirion Technologies (Rados) Oy, Finland. In this study, two designs of individual dosemeters were considered: the TL detector MTS-N-type in the form of solid pellets with a diameter 4,5mm, thickness 0,9mm for whole body Hp(10) and 0,7mm for extremities Hp(0.07) measurements.

The exposures of most of these radiation workers remain at natural radiation background levels. Almost 90% of all occupational doses are on the level under 1mSv/quarter. Less than 10 percent are measurements from the range 1-5 mSv. Doses with a limited percentage exceeding the recommended limits are less than 2%. The highest doses measured by the laboratory for whole body and finger rings dosimeters will be presented along with a case study.

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# Alanine detectors as a potential tool for mailed dosimetry in proton radiotherapy

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**Introduction.** Quality assurance of therapeutic photon beams in the form of mailed dosimetry intercomparisons or postal dose audits is widely used in photon radiotherapy centres. On the other hand, no commonly used, standardized dosimetry auditing program has been established for proton centres so far. This situation motivated EURADOS Working Group 9 (WG9) to investigate selected passive detectors in therapeutic proton beams in terms of their application for routine dose audits. EPR/alanine dosimetry system developed at Cyclotron Centre Bronowice (CCB) in The Henryk Niewodniczanski Institute of Nuclear Physics, Polish Academy of Sciences (IFJ PAN) was included in the WG9 research[1].

In alanine exposed to ionizing radiation some stable free radicals are generated, the concentration of which can be evaluated by Electron Paramagnetic Resonance (EPR) spectrometry. Alanine is tissue-equivalent, demonstrates linear response over a large dose range (0.5-500 Gy) and high signal stability. Moreover its read-out is non-destructive. These features make alanine a promising candidate for dosimetry intercomparisons and clinical audits.

**Material and methods.** EPR/alanine dosimetry system developed at CCB is based on Bruker ESP 300 spectrometer and pellet-shaped alanine detectors, of 3 mm thickness and 4.8 mm diameter. The system was used both for dosimetry intercomparison of ocular radiotherapy facilities (passive proton beams) and for dosimetry intercomparison of treatment units working with spot scanning proton beams (active beams)[2].

**Results and conclusion.** The obtained results showed good agreement between measured and planned dose. Moreover, the properties of alanine, such as dosimeter repeatability, batch reproducibility, low energy dependence, low fading and relative resistance to conditions during transport indicated validity of its use in intercomparisons and postal dose audits for proton beams[1].

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# COMPARISON OF OPTIMIZATION TECHNIQUES FOR THE MONTE CARLO SIMULATION OF CLEARANCE MONITORS

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**Keywords:**
Monte Carlo simulations, simulation optimization, clearance monitors for decommissioning

**Abstract:**
In this work, simulations of the clearance monitor HWM-1800 (Ludlum GmbH, Hamburg) with the Monte Carlo (MC) code FLUKA were performed and optimized. Nowadays, clearance monitors are being widely implemented for decommissioning activities in nuclear reactors and particle accelerator facilities. For every situation, a different calibration is typically needed, depending on the material composition of the samples to be released, their volume and the expected radionuclides. For this reason, MC simulations present a practical tool to obtain information about the capabilities of clearance monitors for different configurations.

Early models of clearance monitors worked as counters, providing an energy-integrated value of the net count rates measured. Modern devices have a multi-channel analyzer, which allows constructing pulse-height spectra. The results of the MC simulations were optimized for both of the previously described situations: comparing the integrated net count rates, and performing a spectral analysis. The performance of both methodologies is evaluated in terms of the discrepancy with the experimentally measured values, and the possible improvements for the optimization with spectral analysis are further discussed.

# ICRP’s revised bioassay data and dose coefficients for occupational tritium intakes

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The Internal Dosimetry Working Group of EURADOS has set up a task group to produce a report providing guidance on the ICRP’s revised dose coefficients and bioassay data following occupational intakes of radionuclides (OIR) [1]. The report will provide guidance for individual radionuclides, including tritium, which is examined here.

The updated biokinetic models in the OIR for inorganic tritium and biogenic tritium share a common structure. Following absorption, 100% of inorganic tritium is directly transferred to blood but biogenic tritium is partitioned equally to blood and an organically bound tritium compartment. The updated biokinetic model is based on recycling of tritium between compartments. The longer-term retention of a small proportion of tritium is modelled with a second organically bound compartment from which tritium is removed with a half-time of 365 d.

The OIR provides guidance on allocating particulate tritides to lung absorption Types. Gaseous forms of tritium, namely HTO, HT and $CH_4$, are now allocated to lung absorption Type V, with the amount deposited dependent on the chemical form. Biogenic organic compounds are assumed to have Type F lung absorption.

As the OIR model expresses excretion rates and dose per unit urinary content in terms of activity per day, the activity concentrations that would usually be measured must be normalised to a daily amount. This could be done using values for the reference daily urinary excretion of 1.2 and 1.6 L for females and males respectively (ICRP 89). However, for the purpose of maintaining consistency with previous assessments, the average value (1.4 L), which also corresponds to the ICRP 23 value, could be applied when calculating doses to both males and females.

The dose coefficient for HTO in the OIR, $2 \times 10^{-11}$ $Sv.Bq^{-1}$, is 10% higher than in ICRP 119. The doses per unit urinary content in the OIR and calculated using ICRP 119 are within 10% up to 35 days after intake.

Following intakes of inorganic tritium, discrepancies between the expected and observed excretion half-times can be corrected by modifying the transfer rate between blood and excreta. The direct dose method[2] should still provide a good measure of effective dose, although some modification of the SEE value is required due to the averaging of doses over the male and female and some correction is required due to the difference in doses between tissues.

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# Effect of Successful Remediation on Radon Personal Dosimetry Results

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UK Health Security Agency, UKHSA (formerly Public Health England) operates a radon personal dosimetry service using PADC (CR-39) dosemeters that measure the time-integrated radon gas concentration. The service relies on the provision by the radiation protection adviser (expert) of a workplace-specific equilibrium factor and on the compensation, by means of control dosemeters, for the exposure the dosemeters receive off-shift. In radon-prone areas the latter can be significant. The service is chiefly used in workplaces such as non-coal mines and tourist caves, where radon remediation is costly, and workers are likely to be “classified” (category A). Demand has more than doubled in the last three years to over 1,900 personal dosemeters a year. This increase is likely to be a result of the publication of ICRP report 137, which recommends an increase in the coefficient of committed effective dose per unit radon exposure. This has yet to be implemented in the UK.

Whilst increasing their occupational dose monitoring, employers are also pursuing radon remediation measures. We report here on a strikingly successful example. Stoke Hill Mine is a drift mine operated since the mid-1500s producing Stoke Ground Bath stone for building and carving. In recent years UKHSA took on the Radiation Protection Adviser role for the mine and, in collaboration with mine management, discovered that routine radon levels were relatively high. Management immediately investigated remediation methods and instituted personal dosimetry using the UKHSA service. Between 15 and 20 workers are monitored.

In January 2019 the mine switched to a positive-pressure ventilation system, resulting in a marked reduction in staff radon doses. Using the current (ICRP 65) dose conversion, mean monthly doses prior to remediation were in excess of 1 mSv. In the months following remediation up to September 2019, all but one of the recorded doses were below the detection limit of E50 = 0.15 mSv, with the exception being 0.19 mSv.

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## Keywords:

Radon, Personal Dosimetry, Remediation

# Aircrew is the most exposed group of radiation workers in Finland despite COVID-19 pandemic

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Aircrew members are the highest exposed group of radiation workers in Finland. This study represents the results of individual monitoring of aircrew members and trends in aircrew doses. Factors affecting radiation exposure of workers in aviation and effect of COVID-19 pandemic to the radiation doses of the aircrew members are described.

Airline companies calculate doses of their workers annually with computer program (CARI-6 or CARI-7) and deliver them electronically to the national dose register containing aircrew dose data since year 2001. For this study, the statistics of radiation exposure of aircraft crews during years 2001-2021, were collected from the national dose register.

In aviation, both the collective dose and the doses of individual workers have increased over the years. The magnitude of radiation exposure depends on flight time, altitude, route and on periodic fluctuations in solar activity. The exposure of the aircrew members can be reduced mainly by limiting working hours and taking into account the flying routes in shift arrangements. In Finland, the airlines have restricted the exposure of their aircrew to less than 6 mSv per year.

Before the COVID-19 pandemic, during year 2019, aircrew (1,306 pilots and 3,292 members of cabin crew) represented 29.5% of the 15,600 workers monitored for radiation exposure in Finland, but their collective effective dose of 13.6 manSv constituted vast majority (84.5%) of the total collective effective dose of 16.1 manSv for all workers.

During year 2020 and the COVID-19 pandemic, the number of aircrew workers decreased only slightly (1,289 pilots and 3,070 members of cabin crew) and they represented 29.6% of the 14,700 workers monitored for radiation exposure. However, their collective effective dose decreased sharply to 4.13 manSv. Despite this, their collective effective dose constituted most (69.8%) of the total collective effective dose of 5.92 manSv for all radiation workers. Statistics from year 2021 will show how quickly their collective effective dose can reach the level obtained before the COVID-19 pandemic.

The increased doses before the COVID-19 pandemic can mainly be explained by the changes in the Finnish aviation. In recent years, new routes have been opened from Finland to the Far East and the number of long-haul flights has increased in general. The fleet used by airlines has been replaced with new aircraft flying higher than the old ones. Also, the northern location of Finland contributes to the higher exposure to the Finnish aircrew when compared to most countries in the EU, as the magnetic field of earth gives less protection near the polar area than the equator.

After the COVID-19 pandemic, it is expected that the number of flights and thus the doses of the aircrew members will again increase very rapidly. Future challenges include development of new ultra-long range aircraft that can fly at high altitudes longer routes (over 15 hours) which may also lead to considerable dose increments. Future challenges may be managed by seeking for synergy effects between radiation protection, flight safety and airline business needs.

# TIMEPIX3 AS SPECTROMETRIC AND TIME RESOLVED DETECTOR FOR DOSIMETRY IN PULSED X-RAY FIELDS

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**Keywords:**
Dosimetry, Spectrometry, Pulsed Radiation

**Abstract:**
Hybrid pixel detectors, such as Timepix3, designed for spectrometric, time and spatially resolved measurements, are ideal tools for a detailed investigation of radiation fields. Dead time free readout is possible in the data-driven mode, which allows their use in pulsed radiation fields. Timepix3 additionally comes with the advantage of having a small detector size (each pixel) for high flux dosimetry while being reasonably large (the sum of all the pixels) resulting in small statistical errors.

The PTB reference field for pulsed x-ray radiation according to ISO/TS 18090, is approved for tests of dosemeters. Any relevant parameter of radiation production (tube current, tube voltage and pulse duration) is independently adjustable. Pulses down to 200 µs duration with dose rates up to 4 kGy/h are possible. The RQR (IEC 61267) and N series X-ray qualities (according to ISO 4037-1) realized up to 120 kV, are used for the presented measurements. All results are traceable to the primary standards of PTB.

The behavior and suitability of a 300 µm silicon and a 1 mm CdTe Timepix3 detector in such fields are tested. Results are compared to reference detectors like the PTW 786 ionization chamber and a Hamamatsu 3590 PIN-Diode. The upper and lower limits concerning the reliance to dose rate will be presented.

In contrast to earlier measurements with the Timepix, this presentation will focus on the special and improved features of the Timepix3. These features lead to time resolved energy histograms with a time resolution of better than 10 µs for a single pulse measurement.

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# Lineal energy distributions and moments of protons using track structure and condensed history Monte Carlo simulations

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**Aim**: To study the lineal energy Probability Mass Distributions (PMDs), track- and dose-average lineal energy ($y_F$ and $y_D$, respectively) for mono-energetic protons using Track Structure (TS) and Condensed History (CH) Monte Carlo simulations.
**Methods**: CH simulations were performed with TOPAS version 3.7. A walled Tissue Equivalent Proportional Counter (TEPC) filled with tissue equivalent gas, with the density required to mimic a water sphere of 1.0 µm diameter, was implemented in the simulations [1]. Default electromagnetic and hadronic physics modules were used. The TEPC was placed in vacuum and exposed to a parallel beam of mono-energetic protons with energies between 20 and 500 MeV/u. The specific energy at the sensitive volume was tallied for each event. TS simulations were performed with Geant4 version 10.6 using the Geant4-DNA (option 2) physics list. Tracks of mono-energetic protons were generated in a water volume having a side half-length equal to the delta electrons maximum range. A point source was placed at the center of the volume and mono-energetic protons with energies between 20 and 90 MeV/u were simulated. A track sampling algorithm was used in order to calculate the specific energy for each track in spheres of 1.0 µm diameter [2].
**Results**: The PMDs obtained from CH simulations are characterized by a sharp peak, whose position depends on the primary proton energy, and by the presence of a low probability tail extending above 100 keV/µm. For TS simulations, the PMDs exhibit a wide peak, which broadens towards the low lineal energy region as the energy of the protons increase, and a fall at approximately 10 keV/µm, regardless of the proton energy. For CH simulations, $y_F$ and $y_D$ ranges, respectively, from 3.39 to 0.25 keV/µm and from 4.84 to 1.98 keV/µm. In the case TS simulations, $y_F$ and $y_D$ ranges, respectively, from 2.16 to 0.88 keV/µm and from 3.54 to 2.21 keV/µm. The observed differences can be attributed to the wall effects and to the different physics models. However, their relative importance needs to be assessed.
**Conclusions**: Lineal energy PMDs, $y_F$ and $y_D$ of mono-energetic protons were obtained by means of TS and CH Monte Carlo simulations. Our findings show that TS and CH simulations result on distinct PMDs and different $y_F$ and $y_D$ values. The sources of the observed differences are currently under investigation.

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# External dose measurements at the affected area in Fukushima prefecture to investigate the body-size dependence of the readings of personal dosimeters

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A personal dosimeter (PD) is used for the individual monitoring for external exposure. Issues such as the influence of the subject body size on the PD response and the relationship between the PD readings and effective dose have to be carefully evaluated for the PD measurements of residents living in areas widely contaminated by radionuclides. Dose measurements using anthropometric phantoms in the areas contaminated by the Chernobyl nuclear accident showed that the body size influenced conversion coefficients from ambient dose equivalent (ADE) rates to effective dose [1]. The Monte Carlo calculations using pediatric and adult computational phantoms suggested that the PD readings provided a good estimate as a measurable quantity for the effective dose when radioactive cesium was widely distributed in the ground [2]. In this study, we investigated the body-size dependence of the PD readings through the external dose measurements using three types of electronic PDs (D-shuttle, Chiyoda Technol., Tokyo, Japan; DOSE e nano and DOSE i, Fuji Electric, Tokyo Japan) attached to the front surface of three anthropometric phantoms (5y, 10y and Adult Male) made of human-tissue equivalent materials in open field at the affected area in Fukushima Prefecture. ADE rates were also measured at the height of 1 m by a NaI (Tl) scintillation survey meter. The ratios of the D-shuttle readings to the ADEs were 0.79, 0.77, and 0.72 for the 5y, 10y and adult Male phantoms, respectively. Further results and discussions on the reliability of the individual external dose measurements using the three PDs will be presented in the upcoming workshop.

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# Preliminary Study on Dosimetric Issues for Triage for Criticality Accident – Lessons from Dose Assessment of the JCO accident –

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In a criticality accident, external dose assessment due to neutrons and gamma rays is required for triage of highly exposed personnel. The triage for acute radiation syndrome, highly exposed personnel requiring life-saving treatment should be immediately identified by means of rapid dose assessment. Triage methods for identifying personnel receiving high doses due to neutrons were well evaluated in the experimental manner [1].

No criticality accident has happened since the JCO accident in Japan took place in 1999 [2],[3]. In this work, we reviewed the dose assessments of the JCO employees, and the radiation workers involved in the JCO criticality accident. The criterion of dose for providing immediate emergency medical treatment is greater than 0.5 Gy and three personnel were identified significant exposure in the JCO accident [2]. In addition, dose assessments of the JCO employees and the radiation worker were carried out by means of three different approaches, 1) dose assessment by the activity concentration of $^{24}$Na by means of the whole body counting, 2) direct readings from the personal dosemeters worn, and 3) calculation by coupled with temporal and spatial dose distributions in the JCO site and workers' behaviour records [2],[4].

Treated the JCO accident as a case study of external dose assessment in a criticality accident, we estimated neutron spectra observed in the accident and the amounts of activation products employing the Monte Carlo (MC) method to improve and contribute the triage method for immediate identification and dosimetry of exposed personnel in a criticality accident. In the study, a criticality reproducing uranyl nitrate solution in the precipitation tank as a source was simulated from the K-code calculations using the MCNP6 to estimate the fission yields and initial neutron spectra. The ambient dose equivalent rate by the calculated spectrum was derived 3702 μSv/h at the distance of 75 m in direction of west of the source and found consistent with measured data (3450 μSv/h). The activation products were assessed using the D-chain of the PHITS code by introducing calculated neutron spectra. Production of beta emitting $^{28}$Al was found predominant on the concrete wall of the facility. The comparison of derived doses by the MC calculations with those obtained by the conventional method and the activation products containing personal belongings and surroundings contributing dosimetry will be discussed in detail.

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# The impact of the calibration procedure in proton therapy on the Stopping Power Ratio value for non tissue-equivalent 3D printed materials

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In the planning of proton therapy, it is essential to know the exact range of beam penetration - and thus stopping power - in the various tissues and materials used during irradiation. Any material placed in the path of accelerated protons interacts with them, slowing down and scattering particles as a result of collisions and nuclear reactions. The impact of the new material on the modification of the proton beam is determined by the Stopping Power Ratio (SPR). The aim of this work was to experimentally verify the SPR of the most popular thermoplastic materials for 3D printing using two calibration algorithms - a clinically used stoichiometric calibration curve and Single Energy Computed Tomography (SECT), implemented in Treatment Planning System (TPS), and an algorithm based on Dual Energy Computed Tomography (DECT) scans [1].

Three commercially available thermoplastic materials for 3D printing – Acrylonitrilebutadiene Styrene (ABS), Polylactic Acid (PLA) and Polyethylene Terephthalate Glycol (PET-G) - were selected and six inserts for CIRS EDP (Electron Density Phantom, Model 062M, CIRS Inc.) phantom were printed - two for each material, applying 0.4 and 0.8 mm printing nozzles. Using the two different calibration methods, the SPR parameter was determined for the printed samples and the correctness of calculations was validated experimentally. The algorithm, proposed by Saito and Sagara, was implemented and adapted to the calibration phantom at the Somatom Definition AS tomographic scanner operating in the DECT mode at the Cyclotron Centre Bronowice (CCB) IFJ PAN in Krakow. Calculations of the SPR parameter of the printed inserts were validated experimentally on the dedicated pencil beam scanning (PBS) nozzle at the CCB. Dose depth distribution dosimetry was performed in the Giraffe multi-layer ionization chamber (MLIC) (IBA dosimetry, Schwarzenbruck, Germany).

The SPR values determined on the basis of SECT calibration were underestimated compared to the measurements by ± 5.7-8.7% for PLA and PET-G except ABS 0.4 which agreed to ± 1.6%. The use of the algorithm based on DECT scans allowed to reduce the underestimation and obtain compliance with the experimental results at a level of ± 3.4%.The results are confirmed by literature reports - the use of DECT-based calibration in proton therapy tends to improve the accuracy of SPR estimation for both tissues and non-tissue-equivalent materials.

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# Development of a BeOSL based beta ring extremity dosemeter

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There are no $^{85}$Kr capable beta dosemeters based on the OSL of BeO ceramic. This is a significant drawback for the BeO based systems as it means that beta extremity dosimetry must be performed by using other means, leading to higher complexity and costs at the IMS in question. Thus, following the presentation of BeOSL finger ring dosemeter for measuring $H_p(0,07)$[1], Mirion Technologies (AWST) started developing a BeO dosemeter capable of measuring $^{85}$Kr too.

The primary issues with using BeO or similar for beta dosimetry is the interaction volume of low energy betas, that is $^{85}$Kr, in combination with OSL efficiency for low energy beta particles[2]. Combination of these two effects means that the OSL signal from $^{85}$Kr is significantly lower than for comparable dose due to photon irradiation, or even $^{90}$Sr/$^{90}$Y, leading to a significant under response.

Here, we present newly developed BeO detector with capability to measure $^{85}$Kr, $^{90}$Sr/$^{90}$Y, as well as photons. By optimising detector design and the readout process we have produced the first beta BeOSL prototype for use in a beta finger ring dosemeter capable of fulfilling IEC 62384 criteria for $H_p(0,07)$ extremity dosemeters for radiation qualities N-15+, $^{137}$Cs, $^{60}$Co, as well as $^{85}$Kr and $^{90}$Sr/$^{90}$Y for 0 and 60°.

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# Photon spectra in NPL standard monoenergetic neutron fields

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The photon spectra in a number of monoenergetic accelerator-based neutron fields have been measured using a High Purity Germanium detector.

Standard neutron fields are invariably accompanied by a photon component due to the neutron generating reactions and secondary neutron interactions in the surrounding environment. An increasing number of instruments are capable of measuring both neutron and photon dose, however the neutron-sensitive element may be sensitive to photons, and vice versa. Accurate knowledge of the photon component is therefore required when calibrating and type testing such devices.

A characterized High Purity Germanium detector [1] was used to measure the photon energy spectra from the monoenergetic neutron fields produced using the Van de Graaff accelerator at NPL (144, 250 and 565 keV from protons on a LiF target; 2 MeV from the pT reaction; 5 MeV from the DD reaction; 16.5 MeV from the DT reaction). The measured pulse-height spectra were unfolded to derive the true photon spectra using a response matrix obtained from Monte Carlo modelling. For each spectrum the observed photons were classified according to their origin (i.e. direct photons from the target layer, room-scattered photons from the target, neutron-induced photons from nuclear reactions in the room, and natural background). The photon spectra are presented and discussed. The new spectra were used to improve the photon to neutron dose equivalent ratios from some earlier work at NPL with GM tubes and EPDs [2].

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# Development and Validation of an Internal Dosimetric Analyser to Assist Confirmatory and Routine Radiobioassay

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The IAEA Radiation Safety Technical Services Laboratory has developed an Internal Dosimetric Analyser (IDA) software tool to facilitate the access to dosimetric data and perform calculations related to individual monitoring for intakes of radionuclides and occupational radiation protection. Direct ("*in vivo*") and indirect ("*in vitro*") radiobioassay methods have been developed to detect radionuclides of interest in tissues or organs of the human body or in excreta. IDA serves to correlate measurement data from confirmatory and routine internal monitoring with data obtained from the Occupational Intakes of Radionuclides (OIR) series of recommendations published by the International Commission on Radiological Protection (ICRP) [1 − 4]. The software tool was designed with an intuitive user interface using Microsoft Excel and was written using the programming language Visual Basic for Applications (VBA). The objective of IDA is to keep the internal dosimetry data in the background and allow the dosimetrist to make the necessary calculations so as to be able to decide, given the bioassay method and monitoring period: (i) whether the method and period are appropriate for routine or confirmatory monitoring; (ii) whether the method and monitoring period allow the recording level to be detected; (iii) whether previous intakes are interfering with the current measurement; and (iv) how do uncertainties in the measurement affect the dose assessment? IDA can present further information such as the minimum detectable dose as a function of the time after intake, the derived recording levels for radionuclide mixtures, committed effective dose calculations for data from air monitoring and the ISO 27048 procedure for the assessment of doses based on bioassay measurements. Isodose curves are shown that allow a quick estimate of the committed effective dose when the time of the intake is known. IDA was successfully validated for functionality according to the requirements of ISO/IEC 17025.

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# The development of Fluorescent Nuclear Track Detectors for measurements of personal neutron doses

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The newly developed LiF-based fluorescent nuclear track detectors (FNTD) offer an effective method of measurements of personal neutron doses using the albedo technique. The nuclear reaction of thermal neutrons with $^{6}Li$ isotopes creates two densely ionizing particles: helium and tritium nuclei. Each track produced by such particles in LiF crystals can be easily detected and imaged with a fluorescent microscope [1,2].
In order to test the feasibility of this approach, FNTDs in form of 4×4×1 mm$^3$ transparent $^{nat}LiF$ crystals, were exposed on a PMMA slab phantom to various neutron doses from a bare Pu-Be source [3]. The microscopic analysis revealed a track density of about 145 tracks/1mSv/mm$^2$. A very specific shape of these tracks makes them well distinguishable from any other tracks or artefacts, which results in the background basically equal to zero. This means that a very low detection limit may be achieved, providing a sufficiently large crystal area is scanned. To further improve the detection efficiency we tested the possibility of using $^{6}LiF$ crystals grown with the micro-pulling-down method instead of $^{nat}LiF$ crystals.
LiF FNTDs enable very good discrimination between neutron and gamma doses. The neutron-induced tracks are well visible even after irradiation of a crystal with 1 Gy of gamma radiation.
Our studies also focus on the influence of a temperature treatment of the crystals on the photoluminescence intensity and spectrum shape. The temperature effects were observed at each stage of the process (before irradiation, during irradiation, after irradiation and during PL measurement). Usage of the heat treatment seems to be a feasible way to improve the signal to noise ratio of the microscopic images.

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# Using single-cell DNA sequencing as a dosimetric tool- An exploratory study

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**Purpose:** Radiation-associated tumours show mutation signatures in their genome, which distinguish them from radiation naïve tumours [1]. Our goal is to find if we can identify similar signatures, well in advance, before radiation exposed cells transform into a tumour. We are also investigating if our technique can be used as a dosimetric tool to estimate the radiation quality and dose deposited in an individual during the event of an incident or an aerospace mission.

We know that stochastic interactions of radiation introduce damage and mutations that are unique to each individual cell's genome. But conventional genome sequencing [2] methods such as bulk cell sequencing cannot detect such unique mutations. Therefore, we are examining if single-cell DNA sequencing may be used to reveal the mutational effects of ionizing radiation in exposed cells. We expect to see different mutation pattern for high- and low-LET radiation and thus we hope to discern the radiation quality from genomic mutations in cells.

**Methods:** In this work, four identical samples of a human B-lymphoblastoid cell line were irradiated *in vitro* using 6 MV X-rays from a medical linear accelerator. They were exposed to sham irradiation (control), 0.5 Gy, 1.5 Gy and 3 Gy respectively at a common dose rate. Irradiated samples were incubated for 24 hrs, and then DNA was extracted from approximately 500 cells per sample and subsequently subjected to single-cell whole-genome DNA sequencing technology. The well-characterized genome of our B-lymphoblastoid cell line was used to establish the baseline mutations in our control and to identify radiation-induced mutations in the three other samples. Copy number alterations (CNA) were identified and examined in individual sequence data from all four sample groups. Radiation-induced copy number (CN) gains and losses were counted.

**Results:** We observed a dose-dependent increase in the number of CNA in our sample groups, where the number of CN losses increased significantly with radiation dose. We also observed a dose dependence for the size of the chromosomal aberrations.

**Conclusions**: Our findings suggest that single-cell sequencing techniques may be used to directly examine the mutational effects of ionization radiation in human cells. We are presently working on reproducing these results with repeated experiments. If confirmed, we posit that our strategy of examining DNA anomalies will open up new avenues for radiation biodosimetry.

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# Measurements of the Recombination Index of Radiation Quality in photon radiotherapy

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The concept of radiation quality is closely related to the effects of the interaction of ionizing radiation within a given tissue material, and thus the relative biological effectiveness. The quality factor can be experimentally measured by using recombination methods and chambers. It has been shown that a basic quantity called the Recombination Index of Radiation Quality, here denoted by $Q_4$, approximates well the Q(L) function, as defined in the next ICRP Reports [1]. In practice, $Q_4$ measurements are most often performed with large, cylindrical chambers used for detecting or monitoring ambient dose equivalent H*(10). Due to specific features of recombination chambers, they are especially suitable for performing specific measurements in complex mixed radiation fields of a wide range of particle types and generated by the beam of pulsed structure. Such radiation fields are created during medical procedures with high energy ionizing radiation (especially in radiotherapy). The detailed characteristics of the radiation fields (mainly by experimentally measured values of quality factor) in the case of radiotherapeutic procedures are important in terms of dosimetry of patients, as well as the medical staff, and in some cases, also members of the public.
The aim of the study was to present results of $Q_4$ measurements with the use of recombination chambers during radiotherapeutic procedures in the vicinity of the medical linear accelerator.
All measurements were carried out in the Maria Sklodowska-Curie National Research Institute of Oncology Krakow Branch, Krakow, Poland, with the use of Clinac 2300 C/D medical linear accelerator (Varian) during the implementation of real radiotherapeutic plans. In order to assess $Q_4$ values, seven radiotherapeutic plans were prepared in 3 various techniques (3D-CRT, IMRT, VMAT) and various combinations of 2 different acceleration voltages of high-energy photons (6 MV and 18 MV) for the case of prostate cancer. An anthropomorphic RANDO phantom was used as a surrogate of the patient's body. Recombination chambers were located on the therapeutic table, 100 cm from the beam axis at the height of the isocenter. Measured values of $Q_4$ (also $(Q_4)_n$ for photoneutrons contribution) will be presented and discussed in the context of the influence of the choice of the irradiation technique and acceleration voltages of high-energy photons on the mixed radiation field around the medical linear accelerator.

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# Multisignal ionization chamber with a $B_4C$ coating as an active neutron beam spectrometer: Monte Carlo simulations

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Neutron spectrometry, considered as an environmental measurement technique, is widely used in [1]: laboratories for the neutron instruments calibration or characterization, cross-section measurements and physical experiments; nuclear or fusion workplaces, medical/industrial accelerator workplaces; and also cosmic-ray radiation fields. Neutron spectrometry considered as a neutron beam measurement technique is required mainly for the medical neutron beams such as for fast neutron therapy or boron neutron capture therapy [2] and is the major interest of this work. It is focused on an active method for intense neutron beam spectrometry using the ionization chamber.

The construction of a multisignal ionization chamber is based on theoretical calculations and Monte Carlo simulations [3,4]. The chamber is designed as an active instrument for real-time measurements of the neutron spectrum. It has a cylindrical form and consists of seven polypropylene electrodes-moderators of varying thicknesses ranging from 20 mm to 40 mm – a total of 160 mm what corresponds to a 12” Bonner sphere.

This work aims to present further results of simulations combining the precise full-scale model of the multisignal ionization chamber with the boron carbide layers covering the electrodes’ surfaces. Results, i.e. response functions of the chamber for monoenergetic neutrons are the basis for the discussion about the possibility of the implementation of unfolding procedures for real-time analysis of the neutron beam spectrum using a multisignal ionization chamber.

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# Unfolding of $^{241}$AmBe and $^{252}$Cf neutron energy spectra using passive multi-layer neutron spectrometer


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Passive neutron spectrometers using activation foils or thermoluminescent detectors [1,2] are used widely around the world with many different forms. Usually, they consist of a single-moderator sphere/cylinder made of high-density polyethylene and a few layers containing small thermal neutron-sensitive detectors placed uniformly in the sphere at different depths. Systems based on integral detectors avoid dead-time losses especially in pulsed fields, are insensitive to high electromagnetic noise, or can extract the gamma component coming from high photon contamination of the field.

In National Centre for Nuclear Research, the passive multi-layer neutron spectrometer was developed [3]. It has a form of a 12” high-density polyethylene sphere with six vertical slots located every 60$^{\circ}$ around the sphere at three depths corresponding to about 2”, 4” and 6.5” of moderating material. In the central part of the spectrometer, there is additional space for a recombination chamber for the measurements requiring monitoring of the field or extra information about the quality of the radiation. In the case of only passive measurements, the central slot is filled with the material equivalent to the moderator. Thermoluminescent detectors of type MTS-6 and MTS-7 were chosen for the spectrometer.

This work aims to present the process of the data collection and evaluation starting from the measurements performed at calibration facility with isotopic $^{241}$AmBe and $^{252}$Cf reference neutron sources, ending with the unfolding procedure resulting in the differential distributions of neutrons.

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# Application of silicon position-sensitive detector in nanodosimetric studies with the Jet Counter device

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The Jet Counter is an ion counting nanodosimeter that uses gaseous media and single ion counting technique. A sensitive volume with simulated nanometer-size dimension is obtained by a gas expansion in a form of gas jet release from a reservoir by a pulse operated piezoelectric valve with a repetition rate of 1–8 Hz. The size of the equivalent nanometric target is established from the combination of the pressure in the reservoir and the voltage applied to the piezoelectric valve. It is monitored by the transmission of a 1 keV electron beam at the centre of the sensitive volume. The device is capable of measuring single ions produced in targets of an equivalent size ranging from 1 to 10 nm [1] with counting efficiency equal approximately 57% [2].

Although the Jet Counter nanodosimeter was built in the 1990's [3], its development is still an ongoing process. Its technical design is modernized to increase efficiency and allow better characterization of the device itself. Recently, a 2D position-sensitive semiconductor detector has been installed at the Jet Counter and used to study its characteristics in function of the projectile trajectory. The results of this study will be presented along with the detailed description of the experimental setup.

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# On the Monte Carlo simulation of neutron-induced indirect DNA damage to estimate neutron relative biological effectiveness

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The exposure of patients undergoing high-energy radiotherapy to secondary neutron radiation poses a risk for iatrogenic secondary cancer induction. Ionizing radiation, such as neutrons, inflicts damage to the DNA molecule via energy depositions in DNA atoms (direct action) and via the radiolysis of nearby water molecules that then generate chemical species capable of reacting with DNA segments to induce damage (indirect action). Our goal is to elucidate the underlying mechanisms of neutron-induced carcinogenesis by investigating the roles of direct and indirect action in the formation of clustered DNA lesions (especially those containing double-strand breaks or DSBs), which is believed to be a main pathway from which radiation-induced mutagenic consequences emerge. Our research group has recently explored the role of neutron direct action [1], but a similar study on neutron indirect action was outstanding.

**Objectives**: The aims of this project were to (i) estimate the relative biological effectiveness (RBE) of neutrons for inducing DSB-containing clusters (complex DSB clusters) due to the combined effects of direct and indirect action, and (ii) to determine whether such RBE estimation may be used as a measure of neutron carcinogenic risk.

**Methods**: In this work, the existing simulation pipeline of our research group on the direct action of ionizing radiation (built using the TOPAS and TOPAS-nBio frameworks) [1] was extended to incorporate an experimentally-validated implementation of indirect action. Using our in-house geometric DNA model [1] and the updated simulation pipeline, we simulated irradiations of monoenergetic neutrons and reference 250 keV X-rays. The DNA damage yields obtained from these simulations were used to estimate energy-dependent neutron RBEs for inducing complex DSB clusters.

**Results**: Our results show that the majority of neutron-induced DNA damage events are isolated simple lesions due to indirect action, while most clustered lesions are hybrid (direct and indirect action) in nature. Our estimated neutron RBEs for inducing complex DSB clusters were found to fall under previous estimates that only considered direct action, despite their smaller yields of complex DSB clusters. We suspect that this was due to the much higher density of lesions in neutron-induced damage clusters, resulting in more lesions being counted in one cluster and less increase in total cluster counts.

**Conclusions**: Indirect action is an important DNA-damage-inducing mechanism of neutron radiation. Using the yields of complex DSB clusters alone as the biological endpoint (i.e., not considering lesion density) was found to be insufficient for calculating realistic values for neutron RBE and, by extension, for estimating carcinogenic risk.

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# Development of a new bioassay laboratory at QST

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The national system for radiation emergency medicine in Japan was revised based on the lessons learned from the Fukushima Daiichi Nuclear Power Plant accident. This system was further revised in 2019; specifically, the Nuclear Regulation Authority of Japan designated the National Institutes for Quantum Science and Technology (QST) as the national core center for coordinating and guiding four other Advanced Radiation Emergency Medical Support Centers. These centers will be responsible for providing medical treatments of severely exposed or contaminated patients as well as maintaining human resources to respond to nuclear disaster through a wide variety of training courses. Diagnostic dose assessment for patients is also one of the important missions of the centers.

To enhance the capability for dose assessment of patients internally contaminated with actinides, the QST started operation of a new facility named the Dose Assessment Building for Advanced Radiation Emergency Medicine. A new bioassay laboratory was installed at this facility as well as an integrated *in-vivo* counter (Tamakuma et al., this issue). This laboratory has two rooms for pretreatment of excreta samples equipped with ashing, digestion and purification devices, and one analytical room with various measurement devices.

Our current main interest is to establish a rapid and labor-saving bioassay method for actinides that are likely to cause significant internal exposure. In 2017, five workers from a Japanese nuclear facility accidentally inhaled plutonium compounds; this accident was the first Japanese case associated with medical intervention using decorporation agents (Ca/Zn-DTPA) [1]. QST was involved in the medical treatment and dose assessment, and our bioassay laboratory presented the $^{239+240}$Pu and $^{241}$Am data by collecting and analyzing ca. 200 urinary samples. To cope with similar situations in the future, we are attempting both the optimization of a conventional method based on alpha spectrometry and the development of a screening method based on mass spectrometry. It is possible to measure U, Pu and Am isotopes by alpha spectrometry; however, mass spectrometry offers a more feasible, less time-consuming method to measure these long-lived actinide isotopes with low specific activity. The combined use of two mass spectrometry devices (SF-ICP-MS and ICP-MS/MS) allows us to rapidly measure multiple actinides (e.g., $^{234}$U, $^{235}$U, $^{238}$U, $^{237}$Np, $^{239}$Pu, $^{240}$Pu) in small-volume urine samples (e.g., 20 ml) and to present the isotope ratios (e.g., $^{240}$Pu/$^{239}$Pu) for source identification [2]. For fecal samples, microwave-assisted aging and digestion are used to shorten the pretreatment to < 8 h. Our developed techniques have been tested and validated through relevant exercises (e.g., PROCORAD) for the purpose of obtaining future international accreditation.

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# Rapid measurement of actinides in urine by mass and alpha spectrometric methods

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In 2017, five workers were involved in the internal contamination accident at a nuclear facility of the Japan Atomic Energy Agency in Oarai-town, Ibaraki Prefecture, Japan. They accidentally inhaled plutonium compounds (containing Pu and Am isotopes) and the maximum committed effective dose was estimated as a range of 100–200 mSv [1]. Internal contamination with alpha-particle emitting actinides, such as $^{239}$Pu, $^{240}$Pu, is likely to bring a large amount of dose to the tissues of persons even if the intake amount is small. Therefore, after this accident, medical intervention using decorporation agents (Ca/Zn-DTPA) to patients was performed for the first time in Japan. This chelation therapy was continued for several weeks or months and ca. 200 urinary samples were collected from the five workers before and after each DTPA administration. To cope with similar situations in the future and provide timely information for prompt decision-making in radiation emergency therapy, we are attempting both the optimization of a conventional method based on alpha spectrometry and the development of a screening method based on mass spectrometry [2].

In terms of mass spectrometric method, to eliminate matrix, polyatomic interferences and tailing effects from U, after $LaF_3$/$CaF_3$ coprecipitation, Pu isotopes were collected after removing U effectively (exhibiting a high decontamination factor of $10^8$ for $^{238}$U) using a single chromatographic column packed with 2 mL AG MP-1M anion exchange resin. The Pu isotopes were measured using $^{242}$Pu as a yield tracer with yields of 76±5%. Using ICP-MS/MS with low background, very low method detection limits for $^{239}$Pu, $^{240}$Pu, and $^{241}$Pu of 0.057, 0.126, and 76.8 mBq $L^{-1}$, respectively, were obtained for 20 mL of urine sample. These detection limits were comparable to those of SF-ICP-MS with higher sensitivity [2].

In terms of alpha spectrometric method, after decomposing organic matter completely with $HNO_3$ and $H_2O_2$, in order to eliminate matrix and interferences, the combination of the iron hydroxide co-precipitation and the automated separation using the TEVA and DGA resin cartridges was used prior to 48 h counting using alpha spectrometer. This method presented stable and high (~ 80%) yields for Pu, Am, and U, and low method detection limits (< 0.5 mBq using 500 mL urine sample).

Subsequently, these alpha and mass spectrometric methods were applied to participate in the intercomparison organized by the Association for the PROmotion of Quality COntrol in RADiotoxicological Analysis (PROCORAD), France, for method validation.

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# Development of a new integrated in-vivo counter system at the QST

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The National Institutes for Quantum Science and Technology (QST) was designated in 2019 as Japan's National Core Center for leading and coordinating four Advanced Radiation Emergency Medical Centers (i.e., Hirosaki University, Fukushima Medical University, Hiroshima University and Nagasaki University). Their main roles are to develop human resources to maintain radiation emergency preparedness in Japan and to cope with radiological emergency situations associated with medical treatments including dose assessment for patients.

To enhance the capabilities of dose assessment, the QST developed and installed a new in-vivo counter system at our new facility in 2021. This system was designed as an integrated system available for both whole-body and lung counting. In the system, three large-sized HPGe detectors with active volumes of 6,500 mm2 x 30 mm each are mounted above the subject on a bed. These detectors are arrayed end to end in the body height direction when used as a whole-body counter, whereas two of the three detectors are placed over the right and left chest when used as a lung counter. The HPGe detectors are capable of detecting low-energy photons such as X or □ rays emitted from plutonium isotopes as well as moderate-energy photons from typical fission or corrosion products (e.g., 137Cs, 60Co). The system also has a rectangular-shaped NaI(Tl) detector (40.6 cm x 12.7 cm x 7.6 cm) under the bed, which can scan from the head to the foot of the subject or be fixed at a position of concern. The whole of the system is installed in a low-background chamber consisting of 20-cm-thick iron plates. The exterior dimensions of the chamber are 2.4 m x 2.4 m x 2.7 m.

The risk of internal contamination by radiation workers is an important concern in Japan due to the prospect of decommissioning old nuclear facilities in the future. QST personnel have gained important experience in direct measurements following actual accidents such as measurements of emergency workers involved in the 2011 Fukushima accident [1] and the internal contamination accident due to accidental inhalation of Pu compounds in 2017 [2]. These experiences have allowed us to improve the functionality of the new in-vivo counter system. Performance tests and optimization of the detector arrangements for this system are now in progress, and the results will be presented at the conference.

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# Polyethylene filter for the TNF2 thermal neutron facility

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The Standard Thermal Neutron Flux Unit, TNF2, in the Brazilian National Ionizing Radiation Metrology Laboratory (LNMRI/IRD)(1), was built for neutron detectors and survey monitors calibration. At the facility, fluence is achieved by moderation of four $^{241}$Am-Be sources (equally spaced) with 0.6 TBq each, and the geometry has been designed with a graphite core and paraffin/graphite blocks surrounding it. The cube has dimensions of 1.2 m x 1.2 m x 1.2 m. A central chamber with 10 cm x 10 cm x 10 cm size at the facility centre is connected to the outside by a central channel (10 cm x 10 cm x 55 cm).

Due to channel small dimensions, it is not possible to calibrate personal dosimeters for thermal neutrons in terms of Equivalent Personal Dose Hp(d), which must be performed with the 30 cm x 30 cm x 15 cm ISO phantom. A polyethylene filter construction was carried out to allow the external irradiation of personal dosimeters in the ISO phantom.

The polyethylene filter was constructed with 29 stacked discs with diameters ranging from 5 to 34 cm. Different thicknesses were simulated to provide the desired effect. The filter showed satisfactory neutron field homogeneity within the measurement area for fluence reduction and proper distance selection from the cube to the ISO phantom. This new irradiation configuration was also experimentally tested and compared to simulation results with MCNPX.

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# Conversion coefficients from ambient dose equivalent to effective doses of various age groups in areas affected by the 2011 Fukushima nuclear disaster

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In case of nuclear accidents, it is important to determine the magnitude of radiation doses received by people living in contaminated territories. Direct measurement of individual external doses using personal dosimeters (PDs) worn by subjects is the most reliable solution for this necessity. However, it is not practical to expect that all returnees to the affected areas wear PDs. Thus, conversion factors from the ambient dose equivalent (ADEs) to the effective dose for various aged individuals must be determined appropriately. Previous studies demonstrated that the ratio of PDs to ADEs at various areas of Fukushima Prefecture mostly existed in the range of 0.6-0.7 [e.g.,1-3]. These studies provided evidence that PD readings would be reasonable estimators for the effective doses of adult subjects at locations where radioactive cesium is widely and uniformly deposited on the ground. In the present study, we experimentally obtained the response of PDs (D-shuttle, Chiyoda Technol., Tokyo, Japan; Dose i, Fuji Electric, Tokyo, Japan) attached to the front of five age-specific anthropometric phantoms (newborn, 1 y, 5y, 10y and adult male) for lateral radiations of $^{137}$Cs with various rotational angles. The PD/ADE ratio for the D-shuttle on the five phantoms with rotational irradiation simulating the exposure in Fukushima Prefecture were 0.87, 0.85, 0.83, 0.83 and 0.77, respectively. The ratio for newborns was 11 % higher than that for adult males. This body size dependency can be explained by the PD response to exposure direction. We discussed the results of this study and our field experiments to construct conversion coefficients from the ADE to the effective doses for age-group residents living in affected areas, which will be presented at the conference.

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# Stochastic Properties of Radiation Tracks in LiF Fluorescence Nuclear Track Detectors

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Lithium fluoride (LiF) is a well-known luminescent and optical material recently applied to detect and visualize tracks of ionizing particles [1]. It became possible to measure at IFJ PAN under fluorescent microscopy the photoluminescence emission from a single densely ionizing particle in LiF crystals with the resolution of about 0.5 μm. Such tracks were observed following exposure of LiF crystals to ion beams, $\alpha$-particles, neutrons and even electrons and γ-rays [1]. The tracks created by fully ionized high-energy nuclei form straight lines reproducing the actual path of particles in matter. These lines with local minima and maxima are not continuous but broken by places without measurable fluorescent emission (“gaps”). The effect is stronger for low-Z ions characterized by lower stopping power. The physics of this effect is not clear since the average energy deposited by ions is high and, at the resolution of about 0.5 μm, should lead to an uniform fluorescence emission along the track. The gaps are also visible in other track detectors, including FNTD detectors based $Al_2O_3$:Mg.

In this research the fluctuations of energy deposition along the particle tracks were analyzed in order to explain the observed effect of gaps. The optical density distribution along heavy charged particle tracks were measured at LiF crystals. The distributions are partly following energy loss distributions as calculated from of Vavilov formula [2]). Monte Carlo track structure codes were next applied to calculate the stochastic of energy deposition in heavy ion tracks at volume size ranging from a few nanometers to a few micrometers. Monte Carlo particle transport codes were applied to simulate individual ionizations and excitation in the track structure of the heavy ions. The simulations were performed with nanometer resolution in wide range of volume size (from few nanometers up to tens of micrometers)to reveal the stochastic nature of energy depositions patterns. The effect could be partly explained by fluctuations produced by delta-rays traveling outside the direct vicinity of the track.

**Acknowledgments:**
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# OSL properties of common medicines as emergency dosimeters

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In case of a large-scale radiation incident (e.g. a nuclear accident or a terrorist attack with a dirty bomb), thousands of people could be exposed to an unknown amount of ionizing radiation. In such a situation there will be a need for a fast method of assessing the absorbed doses in order to implement the *triage*, i.e. segregation of victims according to the degree of injury and need of immediate treatment. The general public is not equipped with dedicated radiation dose monitors. For that reason, in recent years scientists have been investigating effects induced by radiation in various personal objects [1,2,3], that are kept close to a human body, to use them as so-called emergency dosimeters.

Some of the so-far developed methods (based e.g. on the luminescence of electronic elements) work quite well under laboratory conditions when there is time for laborious preparation and analysis of samples, but their effectiveness is doubtful for real emergencies and field applications. Another disadvantage is that the measurement requires destroying of often valuable items.

As was shown in our previous work, the popular painkillers based on ibuprofen or paracetamol can be potential candidates for emergency dosimeters [4]. The aim of this work was to extend the examination of the OSL properties to other types of common over-the-counter pharmaceuticals. The next group of medicines, which were investigated are cold and gastric medications.

The measurements of such luminescence properties as reproducibility, dose-response, fading and spectrum emission were performed. Preliminary results showed that the samples present OSL signal sufficient for the estimation of the accidental dose.

Pharmaceuticals seem to be free of all disadvantages of other emergency dosimeters: their composition is standardized, sampling is immediate, the unit value is usually negligible. We expect our results should provide a perfect tool for emergency dosimetry.

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# Trenched silicon diodes for thermal neutron detection

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**Background of the study:**

Neutron detection systems are becoming of prime importance worldwide for radiation protection due to the recent development of technologies in medical and industrial purposes such as accelerators and nuclear reactors, which are regularly associated with neutron hazards. These hazardous neutrons require monitoring because of their biological effect, which depends on their energy. Therefore, the neutron dosimetry assessment is an active area of research for radiation protection purposes. Helium-3 proportional counters had remained the ideal choice to monitor the thermal neutrons with detection efficiency >60%. However, with rare availability and the global shortage of Helium-3, a new generation of detector technologies will be highly needed. One of these alternative development detections is using a neutron detector based on semiconductor type.

**Method:**

The designing system involves two trenched silicon sensors, whereas one coated with 6LiF, 1μm thick as a thermal neutron converter layer facing another trench Si diode. This arrangement increases the neutron detection efficiency where the resulting fission products produced (alpha at 2.05 MeV and triton at 2.73 MeV) will be emitted in opposite directions from the converter material. In addition, it allows measuring the coincidence efficiency, which recognises a true hit of neutron detection.

**Results:**

The Geant4 simulation results of this arrangement with supporting the neutron experimental validation results using AmBe source, 1 Ci activity showed a promising result where are the maximum detection efficiency is 2.25% and the coincidence detection efficiency reaches up to 1.3%.

**Conclusion:**

The system showed promising results of thermal neutron detection. Although low resulted efficiency, the system provides outstanding n/γ-ray discrimination, an excellent energy resolution and does not require high operational voltage compared to the other developed techniques.

# A Geant4 simulation study of monitoring the primary and secondary radiation during hadron therapy applications

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**Background of the study:**

There has been a rapid worldwide increase in the number of hadron therapy centres over the last 20 years, which include Proton and Carbon beam usage as a radiation treatment beam rather than conventional X-Ray radiotherapy. This increase is driven by hadron therapy's advantageous dose distribution when compared to treatment with photons. While the prescribed dose is delivered to the tumour with minimal entrance and exit dose to the surrounding healthy structures, one of the concerns particular to hadron therapy is the secondary radiation that is created during treatment. This radiation is created as a result of proton/carbon interactions with human tissue will increase the probability of secondary cancers developing in surrounding tissues. The quantity of the radiation created depends on tissue type, beam energy, and delivery technique.

**Method:**

This research conducts with Geant4 code aims to develop simulation studies of a water phantom, volume 20 ×20 × 40cm$^3$ dedicated for tracking the primary beam and the associated productions of secondary radiation during Proton/Carbon beam interactions inside a water volume. The hadron beams are simulated and tracked in various positions within the phantom, where the sensitive detector moves inside the phantom along Z (beam) direction by 1mm shift in depth to monitor the primary hadron particles interactions, track the Bragg-peak, monitor and assess the secondary neutrons that may produce as a result of beam interactions with water molecules at each depth along the beam-line when delivering the prescribed dose to the targeted cells.

**Results:**

The Geant4 datasets showed, the equivalent radiation dose of neutrons can be reach up to 7mSv/1Gy of delivered treatment dose of proton beams with energy 229MeV and more than 50mSv/1Gy with carbon beams at energy 5.3GeV (441.6MeV/u C ions). Full simulation datasets which will provide a better understanding of the hadron therapy behaviour within a water phantom via monitoring and tracking the beam distribution of the primary hadrons, the produced secondary radiation and the equivalent radiation dose of the secondary radiation per 1Gy of delivered treatment dose will be presented.

# Latitudinal effect on the position of Regener-Pfotzer maximum investigated by balloon flight HEMERA 2019 in Sweden and balloon flights FIK in Czechia

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When primary space radiation particles enter into the atmosphere of the Earth they generate showers of secondary radiation. The intensity of secondary radiation reaches its maximum, called the Regener-Pfotzer maximum, at the altitude of about 15-27 km. The exact position depends on the geomagnetic effective vertical cutoff rigidity, the phase of the solar cycle, and also on the type of detected particles.
Altitude profile of ionization in the atmosphere at higher altitudes (around and above the Regener-Pfotzer maximum) can be measured using detectors flown on stratospheric balloons. However, with some exceptions, the duration of the stratospheric balloon flights are rather short, usually 1-5 hours with fairly rapid ascend and descend, and thus statistical uncertainties of measured data are relatively high.
In September 2019, the flight HEMERA with zero pressure balloon from Kiruna, Sweden was launched. The flight lasted 16.2 hours and the maximum reached altitude was 34.5 km. Cosmic radiation in the atmosphere was measured using different types of detectors – Geiger-Muller tube, Si-diode detector SPACEDOS and scintillation detector AIRDOS-C with inorganic crystal. All instruments were battery operated; data (particles count and deposited energy) were automatically stored to a flash disk every 10 seconds. Throughout the whole flight, temperature and pressure inside the thermally insulated and electromagnetically shielded box with detectors were also recorded. The temperature variation during the flight was almost 20$^{o}$C, therefore the correction of obtained data for temperature dependence was necessary.
The results obtained with HEMERA flight at the location with low effective vertical cutoff rigidity are compared with data measured using the same radiation detectors during several stratospheric balloon flights FIK over the Czech Republic (the location with higher vertical cutoff rigidity) and the latitudinal effect of Regener-Pfotzer maximum is discussed. Experimental data are supplemented also with simulations.

# Development of a chronic external dose estimation system for individuals living in areas affected by the Fukushima nuclear accident

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A huge amount of radioactive material was released into the environment due to the 2011 Fukushima nuclear accident. The contamination (mainly radioactive cesium) caused elevated ambient dose equivalent (ADE) rates around Fukushima Prefecture. Protective measures such as sheltering, and evacuation have been implemented. And the ADE rates in affected areas have decreased as a result of physical decay, weathering, and decontamination over time. It has not been reported that Fukushima residents received serious external exposure in their daily life so far. However, the residents who visit the area where the ADE rate is comparatively high may increase as evacuees returning home and revival of the lifestyle before the accident advance. Therefore, evaluation of Fukushima residents' external dose is one of the concerns.

In these perspectives, we have developed a chronic external dose estimation system for individuals living in affected areas by referring to the scientific knowledge and the monitoring data accumulated after the Fukushima nuclear accident. This system consists of a browser-based external dose calculation program and a database of various values used for the calculation. The external dose is calculated based on an individual action pattern and ADE rate distribution data processed from the airborne survey results [1]. Chronic external dose over the future can be calculated by using prediction models for ADE rate distributions characterized by ecological half-lives of radioactive cesium for land-use [2][3]. The individual action pattern includes information such as moving routes, staying places, elapsed times, etc. The moving route is created using an open-source automatic route search algorithm [4] working on the open licence map data [5]. We verified the system by comparison with the estimated individual external doses and the readings of the personal dosimeters (PDs) worn by volunteers who was living in Fukushima Prefecture with their action records during the PD measurements. The results and discussions on the verification of the system will be presented at the conference.

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# Screening levels of TCS-172 NaI(Tl) survey meters used for direct thyroid measurements of populations in nuclear disasters

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Although nuclear safety regulations in Japan have been reinforced ever since the 2011 Fukushima Daiichi nuclear power plant accident, it is still a great challenge to develop a reliable method to implement population exposure monitoring following a nuclear emergency. Particular attention needs to be paid to the internal thyroid exposure of children from radioiodine. The present study established the screening levels of TCS-172 NaI(Tl) survey meters when used for direct thyroid measurement. These devices are one of those most widely used for measuring ambient dose rates in Japan and are intended to be used for direct thyroid measurements for screening purposes [1]. In this study, age-specific conversion factors to enable estimations of $^{131}$I thyroid contents of exposed subjects from device readings were evaluated based on both experiments and simulations. The age-specific neck phantoms recently developed by IRSN [2] allowed us to introduce reliable conversion factors, which agreed well with those from the simulations using the corresponding mathematical phantoms [3]. From the results, age-specific conversion factors for those 0-5 y, ~10 y, ~15 y and adults were determined to be 20, 25, 30 and 35 $\mu$Sv h$^{-1}$ per kBq for $^{131}$I, respectively, including a safety margin. The screening levels were then derived as follows: 0.2 $\mu$Sv h$^{-1}$ for $\leq$ 7 y, 0.5 $\mu$Sv h$^{-1}$ for 8–17 y and 1.0 $\mu$Sv h$^{-1}$ for $\geq$ 18 y. If the readings are below these levels within one week after the single intake of $^{131}$I via inhalation, the internal thyroid dose to the subject is assured to be lower than a thyroid-equivalent dose of 100 mSv . While $^{132}$I ingrowth from $^{132}$Te incorporated into the body via inhalation simultaneously with $^{131}$I can increase the reading of the device, its contribution is small if the intake occurs 3-4 days or more after the reactor shutdown. A look-up table that converts readings into thyroid-equivalent doses at different times after inhalation was also prepared.

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# Experiments to obtain source efficiency of skin contaminated with solution containing alpha emitters

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In radiation emergency medicine, it is essential to know the quantity of radionuclide(s) causing skin and/or wound contamination to determine the appropriate response. However, there is a great difficulty in this task because of the wide variety of contamination on the human skin. In the accident at Oarai Research and Development Center of Japan's Atomic Energy Agency in 2017, five workers were internally contaminated with plutonium compounds [1]. One important lesson from this accident was that skin contamination with alpha emitters is likely to be missed after decontamination with water is employed, resulting in considerable overestimations using lung counter measurements, due to the remaining skin contaminant.

The present study addressed the source efficiency of contaminants with alpha emitters on the skin, one of the components contributing to the counting efficiency. The source efficiency was experimentally obtained through measurements of pig skin samples with an alpha spectrometer at a fixed geometry. The skin samples were prepared from skin tissue of 6 month-old Landrace pigs on which 100 ml of $^{241}$Am standard solution (nitric acid-based) was painted and dried overnight. The alpha spectrometer had a probe equipped with a passive implanted planar silicon (PIPS) detector with a diameter of 46.5 mm (CAM-1700-AM, Canberra Inc., USA). The alpha spectra obtained from the experiments were analyzed using the AASI code [2] in terms of the peak energy shift and the peak broadening in comparison with the alpha spectrum from an $^{241}$Am electrodeposited source with no self-absorption.

As a result, it was found that the peak energy was lowered by ~0.3 MeV and the peak width (FWHM) was increased by ~1.7 times in the test samples. The source efficiency for the test samples was 0.37 on average, suggesting that about one-quarter of the loaded radionuclides entered hair follicles and or deep sites of furrows in the skin.

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# Experimental and Monte Carlo study of energy response of BeO-based OSL detectors within photon energy range up to 15 MeV

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Response of personal dosemeters to high energy photon radiation is of great interest nowadays due to a spread of new radiation technologies (e.g. use of LINACs in medicine) and expansion of occupational exposure domains (e.g. aircrew and space flight exposure). Recent ICRU95 publication [1] has expanded the range of relevant photon energies from 3 MeV upwards, setting new horizons for individual monitoring.
BeO material is extensively used now with several mass produced and some experimental OSL readers. This material has advantages of excellent OSL properties, simple readout and reasonable energy response in low energy (below 100 keV) range.
This study is designed to investigate energy dependence of OSL response at higher photon energies. Several variants of BeO materials [2] with various dopants were irradiated at LINAC (Varian Linac TrueBeam) with energies of incident electron beam 6, 10 and 15 MeV. Irradiations were performed on a PTW RW3 slab phantom with built-up layer in front of the sample.
The experimental data was used as a benchmark for Monte Carlo calculations performed with MCNP6.2 code. Upon successful comparison of experimental and Monte Carlo data, the calculations were expanded to several other exposure configurations including irradiation in free air and behind various filters (copper, aluminum, tin), for various detector thickness (0.5 to 3 mm) and for monoenergetic photon beams with energies up to 15 MeV.
The findings of this study will add knowledge regarding behavior of existing and potential BeO OSL personal dosemeters in the photon fields within energy range above 3 MeV.

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# Retrospective EPR dosimetry in glass and tooth enamel – results of MUG group in RENEB INTER-LABORATORY COMPARISON 2021

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Electron Paramagnetic Resonance (EPR) is recognized method in retrospective dosimetry, based on detection of stable paramagnetic centres induced by radiation in biological materials, like enamel [1], bones [2], nails [3], mollusc shells [4], or artificial materials, like glasses [5] and other. Recently many studies are focused on potential use of screen glasses from mobile phones. EPR dosimetry in glass has been included in the recent RENEB project as one of the physical dosimetry methods verified in an international intercomparison. EPR group from Medical University of Gdansk (MUG) was one of participants in this intercomparison.

Methods: In the EPR part of this project three types of samples were distributed among participating laboratories: 200 mg samples of mineral glass from hand watches, 100 mg samples of Gorilla glass from smartphones and 5 mg samples of tooth enamel. The glass samples included six calibration samples (0 Gy, 0.5 Gy, 1 Gy, 2 Gy, 3 Gy and 6 Gy) and three samples with unknown doses, to be determined by the participants. Three enamel samples were irradiated with unknown doses - calibration samples for enamel were not provided by the organizers. The irradiations were performed in Bundeswehr Institute of Radiobiology, Munich. The EPR measurements in MUG were performed with Bruker EPR EMX spectrometer in X-band, starting from 8th day after irradiation and for the glass samples were continued next 4 months for studying evolution of the dosimetric signals in time. The enamel samples, in which the dosimetric signal is stable, were measured one week after the irradiation of the samples with the unknown doses. To calibrate the dosimetric signal, the enamel samples were irradiated later with dose 20 Gy by 6 MVp X-rays in MUG and measured again. The determination of the dosimetric, signals was performed by numerical decomposition of the EPR spectra into experimentally obtained background and dosimetric spectral components.

Results. The reconstructed doses in terms of (kerma in air) in the samples were:
Watch glass: -0.05 Gy, 1.03 Gy, 3.16 Gy. Gorilla glass: -0.03 Gy, 1.20 Gy, 3.44 Gy
Enamel: 0.1 Gy, 1.3 Gy, 3.1 Gy. The respective nominal (real) doses, revealed by the organizers after the data were reported, were: 0 Gy, 1.2 Gy, 3.5 Gy.

Conclusion: The differences between the reconstructed and the nominal doses were small. The accuracy was sufficient for triage of people at the level of 1-2 Gy.

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# Evaluation of protective properties of individual shielding used in X-ray diagnostics and therapy

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After the discovery of X-rays, medical imaging became one of its main applications. However, the radiation effects and health risks associated with its use were quickly recognized and had to be minimized. The use of lead and other X-ray attenuating materials became commonplace to protect patients and medical personnel by the early 1920s [1]. By the 1950s, shielding materials allowed radiation dose reductions of up to 95% in general radiography [2]. Developments in the above area continue to this day. The replacement of lead shields with lighter composite and lead-free materials has made the verification of their attenuation properties requiring more advanced testing. This paper presents the results of the lead equivalent verification for various types of individual shields using the new measurement techniques and geometries.

Measurements of the lead equivalent as well as other physical parameters determining the attenuation properties of protective materials were carried out in accordance with the newest recommendations [3]. The measurement method was based on the comparison of the recorded air kerma rate passage of the radiation beam through the tested shields and precisely characterized high-purity reference lead absorbers. Measurements were performed using a constant-potential X-ray source in the range of 60 kV to 150 kV and clinically used X-ray tube filtration. A set of ionization chambers connected to a reference electrometer was used to properly determine the dosimetric quantities. In order to compare the results and to take into account the different contribution of the scattered radiation to the final response a number of measurement geometries were used: narrow beam geometry, broad beam geometry and inverse broad beam geometry. Protective properties were determined for lead and lead-free individual shielding of the trunk, thyroid, gonads, and eyes.

The measurement of the protective properties of the samples confirmed their high efficiency. The tests did not show any mechanical damage to the protective material. The degree of reduction of the X-ray beam is significantly higher for lead-based samples than for lead-free shields in the whole range of high voltages used clinically. On the other hand, Pb-free composite aprons are nearly 30% lighter, which is an advantage compared to heavy lead aprons. The measurements showed few cases in which the measured lead equivalent was lower than declared by the manufacturer.

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# DEVELOPMENT OF PORTABLE THYROID DOSE MONITORING SYSTEM USING GAMMA-RAY SPECTROMETERS

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A portable thyroid dose monitoring system consisting of a couple of spectrometers, a detector shield and an operation software has been developed in order to assess the equivalent dose to the thyroid for workers and members of the public in a high dose rate environment at an early stage after a nuclear accident. The $LaBr_3$(Ce) scintillation and CdZnTe semiconductor detectors were selected as the spectrometers of the system for the public members and the workers, respectively, in the view point of the energy resolution and sensitive volume, which affects the detection efficiency of the detector[1]. The material and design of the shield were optimized by the calculation and experiment. As the size of the shield is limited on upper and lower sides of the throat especially for small children, tungsten heavy alloy, whose large stopping power can achieve a downsize of the shield, was selected as a shielding material of these two sides. Lead was employed as the material for the other part. As varying the arrangement of the subject and the spectrometers affects the accuracy of the dose assessment, positioning jigs were developed to fix the arrangement[2]. A GUI program was developed to control the system and evaluate the thyroid equivalent dose. Their characteristics in a high dose rate environment were studied in the photon calibration fields at the Facility of Radiation Standards of the Japan Atomic Energy Agency. We confirmed that the spectrometers with the shield have enough high energy resolutions and efficiencies to assess the thyroid equivalent dose of 10 mSv in a high dose rate environment up to 20 μSv/h. In this presentation, the performance of the system will be introduced.

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# Dose rate maps of LIEBE target employing Monte Carlo code MCNP

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CERN's MEDICIS facility contributes to medical research by producing novel radioisotopes, elements with too many or too few neutrons to be stable. These radioisotopes not only help diagnose cancers and other diseases, but can also deliver precise radiation doses to treat diseased cells without destroying the surrounding healthy tissue. It is driven by CERN's ISOLDE facility, which directs a high-intensity proton beam from the Proton Synchrotron Booster onto specially developed thick targets. MEDICIS works by placing a second target behind ISOLDE's to produce radioisotopes, which are then extracted via mass separation, implanted in a small foil and delivered to research facilities. In addition, in order to increase the primary beam intensity in the next generation of Radioactive Ion Beam installations, the production of targets capable of dissipating high beam power, especially for molten targets, is a major challenge. A direct molten loop target concept for short-lived isotopes for EURISOL was proposed in that context. Molten metal circulation allows the production of droplets that enhance the diffusion of radioisotopes. The concept also includes a heat exchanger that ensures thermal equilibrium under high proton beam power interaction. To validate this concept in the ISOLDE operational environment, a target prototype, named LIEBE (LIquid Eutectic lead Bismuth for Eurisol), has been designed and assembled.

This paper presents dose rate maps of the LIEBE target. A series of proton transport calculations are performed using the MCNP Monte Carlo code for simulating the proton beam interaction with the target. The latest LIEBE target model has been created in high fidelity geometry and FENDL-3.1 cross-section data library has been used. The obtained dose rate maps in the LIEBE target using weight windows generated using ADVANTG will be presented in the paper.

*Keywords: neutronics, dose rate, LIEBE*

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|---|---|
| **Topic Category** | Neutrons and ions in medicine (dosimetry for proton, ion radiotherapy, BNCT and fast neutron, radiotherapy, patient dosimetry) |
| **Presentation Preference** | ☐ Oral Presentation ■ Poster Presentation |

# Proton beam forming and methods of irradiation on the proton electronic irradiation facility at IFJ PAN in Kraków

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The aim of the work is to prepare the irradiation station for 60 MeV proton beam irradiation of electronic devices and electronics construction materials for radiation hardness. The irradiation station has been located at the Experimental Hall at AIC-144 cyclotron building in the Institute of Nuclear Physics.

A mono-energetic 60 MeV proton beam from the AIC-144 cyclotron is delivered to the experimental hall and is formed on the optical bench. A set of lead, copper and aluminium foils and a set of PMMA plates were used as a reconfigurable scatterer. Protons with selected energies from 10 MeV to 60 MeV are available for the sample irradiation. An automatic, remote controlled 2D scanner will be constructed to enable moving the probes perpendicular to beam's axis.

Two methods of proton beam forming for sample irradiation have been applied. The passive scattering method, which allows irradiation of a wide field with a diameter of up to 12 cm with a uniformity of +/- 15% and a narrow beam irradiation with a diameter from 1 cm to 3 cm and the use of a 2D moving table, that moves the sample during irradiation. The use of the moving table increases the field in which samples can be irradiated with uniform fluence up to the size of 25 x 35 cm. The FLUKA code (v. 4-2.0) has been applied for calculation of the dose and fluence distributions. Spatial dose distribution was determined using the active (ProBimS, LynX) and passive dosimetric methods (Gafchromic films, 2D TL foils).

For small dimensions of radiation fields, depending on the assumed irradiation conditions, the better choice is irradiation with the use of a passively scattered beam, not with a 2D scanner. For larger fields, the use of the beam is more efficient when irradiated with the scanner. The use of the XY scanner allows irradiation with protons of energy > 58 MeV for a field with a diameter greater than 10 mm and homogeneity better than 10%.

# Application of $LiMgPO_4$ crystal for proton beam quality control in radiotherapy


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Wide range of applications of proton beam such as radiotherapy, cosmic industry, radiobiology or medicine requires development of methods which can provide maximization of safety and quality. One of the new method for real time monitoring of the proton beam is based on the measurement of radioluminescence emitted by a luminescence crystal under irradiation. The aim of our work is to determine the dosimetric properties of $LiMgPO_4$ luminescence crystal and to determine the suitability of such detector for the quality control of the proton beam.

The research was carried out on a horizontal proton beam irradiation facility at the Institute of Nuclear Physics Polish Academy of Sciences. The proton beam of energy of 60 MeV was produced by AIC-144 isochronous cyclotron. A small OSL crystal (diameter of 2 mm and the thickness of 1 mm) placed on the end of 15 m optical quartz fiber connected by a set of optical filter to the photomultiplier tube, called PORTOS, was applied [1]. The crystals of $LiMgPO_4$ doped with Tb or Tm were grown by micro-pulling down method at the crystal growth facility at IFJ PAN and then cut into slices. The readouts of radioluminescence were taken in the time intervals from 1 μs to 1 s.

The PORTOS response to the proton beam was investigated in the wide range of energies and currents. The time distribution of macro-pulses structures of the proton beam was verified. For the tested crystals, the dependence of the efficiency on the proton energy was determined. The dose depth distribution measured by PORTOS device has been compared with PTW Markus ionization chamber silicon diode standard used in proton beam QA. The preliminary results showed that the PORTOS device with a $LiMgPO_4$ detector can be an efficient tool in real time proton beam monitoring.


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# Dose assessment to the fetus location using different irradiation geometries of scanning proton beam when treating a brain tumour of the patient

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It is commonly assumed that pregnancy is a general contraindication for radiation therapy. However, in some cases, to maintain the life and health of the patient, a delay of radiation treatment may not be clinically acceptable. On the other hand, health effects to a fetus from radiation exposure depend largely on the radiation dose [1]. Despite the fact that the proton beam scanning (PBS) radiotherapy may significantly reduce the dose delivered to a fetus comparing to photon or proton passive scattering therapy, it is strongly recommended to searching for new approaches of minimizing it even more. For this reason, different irradiation geometries of anthropomorphic phantom using PBS were carefully evaluated.
Several geometries of pregnant patient have been modelled on the gantry facility using Alderson Rando anthropomorphic phantom (head and chest) and RW3 plates (abdomen, uterus and pelvis). Each geometry included the single uniform field of PBS in order to irradiate the target located in the brain. Irradiations have been performed using two different directions of the beam: left lateral and vertex. Each geometry was irradiated with and without a range shifter (RS) placed in the beam path. Four different geometry measurements of out of field doses at the location of the fetus were performed using WENDI detector as well as FHT192 probe for neutron and gamma dose assessment respectively.
The results show that, the best choice to minimize fetus radiation exposure is the irradiation using lateral beam without RS. For a prescription of 54 Gy in 30 fractions, the total H*(10) to the fetus was estimated to be 58.6 μSv (neutrons H*(10): 35.4 μSv; gamma H*(10): 23.1 μSv). The use of RS with left lateral beam led to an increase in H*(10) to 326.5 μSv (neutrons H*(10): 163.6 μSv; gamma H*(10): 162.9 μSv). The worst case scenario was irradiation of the phantom using vertex beam direction with RS inserted into the beamline, which gave total H*(10)=506.0 μSv (neutrons H*(10): 292.5 μSv; gamma H*(10): 213.5 μSv) measured at the fetus location. Vertex beam irradiation without RS resulted in the total H*(10) equal to 432.9 μSv (neutrons H*(10): 162.0 μSv; gamma H*(10): 270.9 μSv).
Similar to [2] it can be concluded that pregnant patients with a brain tumors could be treated with PBS with acceptable risk (< 50 mSv) using any of proposed geometries. Moreover, fetus dose can be easily decreased by avoiding vertex beam directions or RS in the beam path.

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# Responses of new BSS spectrometers printed with ABS and PLA polymers

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Bonner Sphere Spectrometers (BSS) generally consist of a set of moderating high-density polyethylene (HDPE) spheres with different diameters and a hole at its center, where can be inserted a thermal neutron sensor (active or passive). The use of water spheres or cylindrical and cube-shaped HDPE moderators has also been studied [1].
The filaments of 3D printers are rich in hydrogen and carbon; therefore, have good potential to moderate neutrons. The ABS (acrylonitrile butadiene styrene - $C_8H_8$ $C_4H_6$ $C_3H_3N)_n$) and PLA (polylactic acid - $(C_3H_4O_2)_n$) are popular filament materials used in commercial 3D printers of the FDM type (Fused Deposition Modelling).
In this paper, we used the Monte Carlo code MCNPX version 2.7.0, with ENDF/B-VII.0 nuclear data library, to calculate the response functions of BSS spectrometers with a $^{6}$LiI(Eu) scintillator and HDPE, ABS, and PLA moderators. Simulations were performed for the bare detector and the following sphere diameters, in centimeters: 5.08, 7.62, 10.16, 12.7, 15.24, 20.32, 22.86, 25.4, 30.48, and 38.10. The HDPE responses were compared with other published data. Responses to reference neutron fields (ISO $^{241}$Am-Be, $^{241}$Am-B, and $^{252}$Cf) were also obtained. Ratios of the simulated responses ABS/ HDPE and PLA/ HDPE were calculated. Scanning electron microscopy (SEM), coupled with energy-dispersive X-ray spectroscopy (EDS), was used to investigate the morphology and porosity of the polymers after the printing process, as well as the chemical composition. ABS/ HDPE response ratios for $^{241}$Am-Be neutron field were experimentally obtained for 2” and 3” spheres. Irradiations were carried out with the reference $^{241}$Am-Be neutron source of the Low Scattering Room (LSR) of the Neutron Metrology Laboratory (LN/LNMRI/IRD), in Rio de Janeiro.
Experimental ratios ABS/ HDPE for the 2” and 3” spheres were, respectively, a factor 2.0 and 2.3 lower than simulated ratios. The presence of porosity observed in the printed pieces can influence the experimental results using PLA and ABS spheres. However, the variation of printing parameters such as the thickness of the deposited layer, air space between extrusions, speed and processing temperature, contour angle and extrusion width, as well as post-print chemical treatments allow obtaining pieces with high density and low porosity. The study of these parameters will be carried out in future works.

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# Neutronic activities in IPPLM

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Nuclear fusion is considered as an almost inexhaustible source of energy for future generations. The fusion reactions in plasma are the primary source of fast neutrons. In the result of DD and DT reactions, 2.5-MeV and 14.1-MeV neutrons are produced, respectively. TT reactions can also occur, and neutrons from this process have a continuous spectrum up to 8.8 MeV. Fusion neutrons, which are energy carriers, are indispensable for tritium breeding, but they also modify the properties of materials causing their activation, transmutation and radiation damage. The highly activated materials used to build the fusion power plant may release high dose levels to critical components and cause hazards during the maintenance period. Nuclear characteristics such as decay heat, induced radioactivity and dose rate of activated materials need to be analysed appropriately in the safety analysis. The IPPLM team is involved in studies focused on safety analysis and calculation of decay heat, dose rate and radioactivity for crucial systems of future fusion power plants. The recent results and developments in these studies will be presented.
Moreover, the evaluation of the induced radioactivity of the components is needed to understand the hazards of the radioactive materials. To predict the level of radioactivity of elements irradiated in high neutron fluxes during a long period its possible to use state-of-art codes and nuclear data. To study materials behaviour, impurities concentration of materials, and validate the codes used to calculate activation predictions, the functional materials used in diagnostics and heating systems were irradiated during the experimental campaign in one of the operating fusion facilities. The IPPLM is involved in gamma spectrometry measurements of activated samples using the HPGe detector and comparing numerical predictions and measured quantities. The recent results in this area will be discussed.
The IPPLM has also broad experience in the reconstruction of the neutron spectrum for neutron generators with the application of the different deconvolution methods and activation method [1]. The recent development in this area will be discussed as well.

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# Investigation of different luminescent radiation detectors exposed to mixed radiation fields at CERN-EU reference field facility.

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Within this work, several different types and forms of passive luminescent radiation detectors have been investigated, regarding their response to mixed radiation fields consisting of different nuclear particles. The range of the studied samples included thermally (TL) and optically stimulated detectors (OSL), as well as fluorescent nuclear track detectors (FNTD). The tested detectors were made of differently doped LiF, $LiMgPO_4$ and BeO compounds. The detectors had a form of cold-pressed and sintered pellets (LiF, BeO), but also crystals grown from the melt using novel micro-pulling-down ($LiMgPO_4$) and conventional Czochralski (LiF) techniques. The samples enriched in $^6$Li and $^7$Li isotopes were included in the experiments. All samples were exposed to the mixed radiation fields at the CERN-EU high-energy reference field facility. This radiation field consisted mainly of neutrons of a very broad energy range (from $10^{-8}$ up to $10^3$ MeV). However, other nuclear particles like photons, protons, electrons, pions, muons were also involved in the interactions with investigated materials. The luminescent radiation detectors are sensitive to all kinds of (directly and indirectly) ionizing radiation including energetic ions, but their response strongly depends on the ionization density. The response of differently doped LiF and $LiMgPO_4$ samples to thermal neutrons is quite well known, as several research papers on this topic have been published so far [1-2]. Quantitative and qualitative analysis of radiation field ingredients was possible thanks to the detectors of the different content of $^6$Li and $^7$Li isotopes. Using the photoluminescence (PL) phenomenon in LiF single crystals grown by the Czochralski method, the visualization of individual nuclear particles tracks was also possible [3]. The obtained results will be discussed in detail in the frame of the conference presentation.

**Acknowledgements:**
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# Methodology for dose equivalent assessment in mixed radiation fields combining measurements from different types of detectors.

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Determining out-of-field radiation doses in patients under radiotherapy treatments (photon and proton) is one of the tasks of WG9 (Radiation dosimetry in Radiotherapy) of EURADOS (The European Radiation Dosimetry Group). For this purpose, water slab and anthropomorphic phantoms, filled with several types of passive detectors, are irradiated simulating patient's treatments [1,2]. In this condition, operational quantities defined by ICRP for external irradiation cannot be used for assigning equivalent or effective doses, because some radiation component reaching the organs of interest, as it is the case for neutrons, is originated inside the patient.

The situation is that a mixed radiation field, with each component having a different energy distribution, reaches the points of interest. Dose equivalent at these points cannot be directly measured and must be calculated from radiation quality factors [3]. On the other hand, each type of detector is sensitive to a given energy range of one or several components of this radiation field and is characterized by its response function. Usually, detectors of different types are employed to assign partial dose equivalents due to one type of particles o a given energy range (eg: photon dose, thermal neutron dose, fast neutron dose).

A model is proposed to combine the measurements from a set of different detectors to obtain the total dose equivalent at the measurement point. The model uses Monte Carlo (MC) simulations for transporting from the well-known delivered radiation field to the area where detectors are placed and makes use of either response functions or calibration coefficients of the detectors to calculate the signal recorded.

The main objective of this work is to present this model, which translates into a general equation that could be used whenever a mixed field is present. A dedicated exposure [4] has been performed in a calibration facility to check the validity of the model. Analysis of the detectors irradiated is in progress.

This work has been possible thanks to the Young Scientist EURADOS Grant awarded to one of the authors (MACP).

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# The impact of the COVID-19 pandemic on the dosimetry of aircraft crews in the Czech Republic and Slovakia

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The COVID-19 pandemic has significantly affected many human activities, including air transport. Passenger air transport on scheduled flights in particular has been significantly reduced or eliminated. The restriction was also applied to air taxi services and private flights. In routine dosimetry of aircraft crews, which is done by calculation, these limitations resulted in a reduced collective dose of aircrew, but in some cases an increase in the average annual effective dose due to the increased workload of fewer personnel was observed. However, even during the strict lockdowns, several comparative verification flights were performed with many types of cosmic ray dosimeters. Long-term measurements with Liulin and AIRDOS semiconductor detectors on board Smart Wings and ABS Jets also continued. The paper will present an analysis of routine dosimetry of aircraft crews in the Czech Republic and Slovakia, as well as a comparison of verification measurements on board aircraft with computer program CARI-7.

# Application of gamma spectroscopy to alpha particles dosimetry in BNCT

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Radiotherapy using α particles is becoming an object of increasing interest. It is extremely important to monitor dose deposited by the particles in tissues and therefore – the dose rate during the therapy. Measurements of α particles delivered to the body in the form of radiopharmaceutical or boron carrier as a source of alpha particles in the reaction with neutrons can be only carried out indirectly. Difficulties in the direct *in vivo* measurements of α particles released in tissue are associated with their small ranges, on the order of μm. It is therefore necessary to develop appropriate measurement methods. A detection of low-energy γ photons accompanying the emission of these particles can be a suitable method [1].

Alpha particles are emitted, among the others, in the $^{10}B(n,\alpha)^{7}Li$ reaction. The reaction is the basis of boron neutron capture therapy (BNCT). In this reaction, with 94 % probability, γ photons with an energy of ca. 478 keV are emitted [2].

It will be presented a current state of research to develop a method for the determination of the dose deposited in a tissue-like medium by α particles derived from the $^{10}B(n,\alpha)^{7}Li$ reaction. The proposed method is based on the real time measurment of γ photons produced in the same reaction. A special experimental set up was designed and built to achieve this aim. Measurements were carried out in the field of neutrons emitted from Pu-Be source. Gamma photons were detected by a gamma ray spectrometer based on CdZnTe detector. PADC nuclear track detectors were used to determine the fluence of alpha particles at a given position inside of the boron solution filled phantom irradiated by neutrons. In addition, the simulations based on the MCNP code were performed in order to determine the efficiency of the $^{10}B(n,\alpha)^{7}Li$ reactions measurements [3].

The experimental set up built for the purpose of the measurements carried out in the MARIA Reactor at the National Center for Nuclear Research in Świerk, Poland, will be discussed. The conclusions drawn on the basis of measurement data obtained in this part of the research will also be presented.

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# Fast method of determining the absorbed dose equivalent at a depth of 10 mm H*(10) of gamma-neutron fields based on recombination methods.

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The subject of the work is the presentation of a new measurement algorithm to be used in dosimetry, based on recombination chambers, i.e. pressure gas detectors working in the regime of ion recombination, with particular emphasis on the use of initial recombination. A well-known recombination chamber of the REM-2 type is an ambient dose equivalent detector at a depth of 10 mm H*(10) in fields of unknown composition and a wide energy spectrum of particles. The measurement is carried out by direct measurement of the ambient absorbed dose at a depth of 10 mm D*(10) - measurement of the current $I_S$ for voltage approximately equal to saturation $U_S$ and indirect measurement of the recombination radiation quality factor $Q_R$ by comparing the $I_S$ value with the current value $I_R$ for recombination voltage $U_R$. The recombination voltage is the one for which in the reference gamma radiation field (standard $^{137}$Cs) $\frac{I_R}{I_S} = 1 - 0.01 \cdot R$ for $R = 4$. Providing such detector operating conditions gives the best match of $Q_R$ to the unknown, mixed radiation quality factor radiation fields according to ICRP recommendations (ICRP Publication 60. Ann. ICRP 21) according to the relationship $Q_4 = \frac{1-\frac{I_R}{I_S}}{0.04}$, then $H^*(10) = D^*(10) \cdot Q_4$ [1].

The current measurement algorithm performs the direct current measurement by measuring the voltage at the output of the preamplifier with a feedback loop of known resistance. Such a measurement gives good results and is well described, but its applicability is limited due to the long duration of the measurement and susceptibility to disturbances due to the large time constant of the measurement system. The new measurement algorithm is based on high-frequency measurement of the voltage at the output of the amplifier in the integrator. The time constant of such a system is close to zero and allows to shorten the measurement time dozens of times while improving the quality of measurement through offline analysis.

The new measurement algorithm was tested in gamma and neutron radiation fields with an ambient dose equivalent range from 150 μSv/h to 1 mSv/h. Isotopic radiation sources $^{137}$Cs, $^{60}$Co, $^{239}$PuBe, and $^{252}$Cf were used. The measurement time was shortened from about 10 minutes to about 20 seconds with no decrease in the quality of the measurements compared to the classical method. The results obtained in this way are promising, and further research on the new algorithm is warranted.

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# Dosimetry on basis of common salt (NaCl)

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The study deals with a luminescence dosemeter based on a common, largely available household salt (NaCl). For the purpose of the study, three types of salt were used. The salt detectors had form of pellets made from salt grains of 250 – 400 μm. They were produced using a mould and a hydraulic press with compression force of 3.0 ± 0.5 tons. Triplets of the pellets were inserted into a special holder with filters. For purposes of personal dose equivalent ($H_p(10)$) measurement, the filters covering the individual pellets were made of copper, aluminium and plastic. In case of ambient dose equivalent (H*(10)) measurement, a plastic card containing three pellets was inserted in an aluminium container. Optically stimulated luminescence (OSL) measurements of salt detectors were performed using the Risø TL/OSL reader model DA-20. The analytical protocol used was single-aliquot regenerative-dose (SAR) protocol.

For each type of salt, basic dosimetry characteristics of the detectors were studied. These included dose response of the OSL signal, minimum detectable dose (MDD), homogeneity and fading. Using the special holders, energy response of the particular dosemeters was studied. For this purpose, a wide range of photon energies between 33 and 1250 keV was applied. The personal version of the dosemeter was fixed to the ICRU slab phantom during the irradiations. The environmental dosemeter was irradiated free in the air.

Results indicate that common salt in form of pellets is a very sensitive material. The MDD values obtained for the three types of salt were between 3 and 58 μGy. An inverse fading of the signal was observed. The linearity of the dose response was observed up to 100 mGy for all three types of salt. OSL signal of salt exhibits a stronge dependence on photon energy. As regards H*(10) measurements, it is evident that the environmental dosemeter is suitable rather for higher photon energies (> 100 keV). Hp(10) can be calculated using an appropriate algorithm applied to the OSL responses of the detectors under the particular filters. The potential of salt dosemeters can be realized especially in emergency radiation situations when demands on capacity of monitoring systems can increase. The salt dosemeters represent an easily available and cost-effective possibility.

**References:**

# Human head phantom to ensure dosimetry of dose components for Boron Neutron Capture Therapy made from liquid silica glass.

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Boron Neutron Capture Therapy (BNCT) is a two-stage therapy, offering a chance for patients with head and neck tumors, brain tumors and disseminated tumors. In the first stage of therapy, the patient is administered with a boron compound attached to a carrier which selectively reaches the cancer cell. Then the patient is irradiated with an epithermal neutron beam.

One of the most important problems in the therapeutic process is the correct determination of the four dose components in BNCT: gamma dose, fast neutron dose, nitrogen dose and boron dose. (AEA-TECDOC-1223, 2001) To resolve this issue, the work to prepare an anthropomorphic phantom of the human head is underway.

The aim of the research is to develop a technology of fast production of a human head phantom, prepared based on CT scans and filled with dosimetry gel. The material, used for 3D printing of the phantom, is based on liquid silica glass (Kotz et al., 2017).

Our research is divided into several stages. The first part includes simulations of the phantom at the BNCT facility in the Particle and Heavy Ion Transport code System (PHITS). In the next stage, measurements will be carried out with fabricated phantom prototypes to verify further assumptions of the research concept. The starting point for the obtained results will be a cubic water phantom made of Plexiglas. However, modifications of the prototype will be carried out until the measurement results are as close as possible to the simulations performed (a next few stages).

Through the speech will be presented the premise of the research idea and a summary of the simulations conducted so far.

The measurements described above will be carried out at BNCT facility in National Centre for Nuclear Research. The facility is being built at one of the horizontal channels at the MARIA research reactor and will have a research and training function.

The final stage of the research will be the process of introducing the human head phantom as a clinically applicable product for dosimetry of dose components for Boron Neutron Capture Therapy.

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# Determination of radon exposures in water works located in granitic areas

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Radon in water works located in granitic areas presents a specific problem because larger quantities of natural radionuclides of the uranium and thorium decay chains can be solved in the ground water and therefore also Rn-222, which has a high solubility in water. If the ground water is pumped up to the surface radon can easily emanate, especially if the water is swirled when coming out of a pipe. If the building is additionally bad ventilated very high radon concentrations in air of more than 10 000 Bq/m$^3$ can occur in some cases which can result in high individual exposures of the employees.

The detector system must therefore be able to measure very high radon concentrations. Eurofins Radon Testing Sweden AB has a radon measurement laboratory, which is accredited according to ISO / IEC 17025:2018 [1] and SS-ISO 11665-4:2021 [2] (SWEDAC no. 10243). The accredited measurement range comprises the range from 10 to 8 000 kBq.h/m$^3$. By using a special evaluation software however, the upper limit can be extended to 30 000 kBq.h/m$^3$ which corresponds to a radon concentration of 90 000 Bq/m$^3$ for a 14-days measurement period (336 h). The track recognition system is based on a new algorithm so that tracks can be counted even if they are overlapping. In water works the period for local measurements can be reduced to 14 days because the radon concentration is influenced by the working rhythm of the water work and not by seasonal variations.

Local measurements in waterworks are useful to get an overview over the radon situation in different buildings in order to seize targeted reduction measures. For the determination of individual exposures however personal radon detectors are used, which are worn by the workers during work. This is especially useful if the employees switch back and forth between different buildings of the water production. In contrast to local measurements personal measurements require a low detection limit. A lower limit of 10 kBq.h/m$^3$ corresponds to 0.075 mSv by using a dose conversion coefficient of 7.5 mSv / (MBq.h/m$^3$) recommended by ICRP publication 126 [3]. This lower limit can be achieved by good raw plastic material, a very clean manufacturing process of the ready to use radon detectors and again the track recognition software.

Eurofins passive radon detectors show a high measurement range from 10 to 30 000 kBq.h/m$^3$. They are therefore equally suitable for local measurements of high radon concentrations and individual monitoring of personal exposures in water works.

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# Dosimetry of the MARIA Reactor H2 channel beam in research on boron-neutron capture radiotherapy

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There is a new facility being built for research on boron-neutron capture therapy (BNCT) at the National Center for Nuclear Research, Świerk, Poland. This radiotherapy modality depends on irradiating with a thermal neutron beam a tumor to which the boron ($^{10}B$) compound has been selectively delivered. The collision of a neutron with a boron atom produces α particles with high linear energy transfer (LET) and $^{7}Li$ recoil nuclei. Both molecules move on short distances, not exceeding the size of an average cell (about 10 μm), so their interaction is limited to cells that selectively bind boron. BNCT therapy offers the possibility of improving the therapeutic index of radiotherapy.

In our experiment we have compared results of physical and biological dosimetry of the MARIA Reactor H2 channel beam, and created protocols for the irradiation of biological materials in the emerging channel.

Initial studies of the effect of mixed radiation in MCF-7 breast cancer cell culture were conducted. In order to relate the observed effects to the neutron ratio in the beam, the boron-4-$^{10}B$-borono-L-phenylalanine (BPA) compound at a concentration of 0.925 μM was added to a part of the culture 24 h before the test. After irradiation, the quantification of the survival fraction (SF) and the proliferation capacity of the tumor cells was performed using a clonogenic assay. Correlation of the results of physical dosimetry read from thermoluminescent detectors (TLD) with the results of biological dosimetry has been sought.

MTS-N and MTS-7 TLD detectors were used in physical dosimetry. MTS-N detectors contain the $^{6}Li$ isotope, which has a large cross-section for reactions with neutrons. Concurrently, the MTS-7 detectors have the $^{7}Li$ isotope, which does not react with neutrons. Both of these isotopes have similar cross-sections open to ionization by gamma quanta, which allow to use them simultaneously, separating the radiation components and reading the absorbed doses from both neutron and gamma radiation. A total exposure dose of 2.8 Gy was read, of which 90% was gamma radiation and 10% - neutron radiation.

The survival rates (SF) of cells after irradiation with or without BPA were compared and were 0.84 and 0.58, respectively. Thus, it seems that the share of 10% of neutrons in the beam may be responsible for an additional 26% difference in the mixed beam lethal effect, which is directly related to the collision of a neutron with boron atoms.

The results of biological dosimetry correlate with the results of physical dosimetry in terms of the visible effect of BNCT dependent on the admixture of pure neutron radiation in the beam and the presence of the boron carrier in the biological system. Biological dosimetry can be used to extrapolate the physical conditions of the beam to its biological effectiveness and thus to the clinical conditions of BNCT radiotherapy.

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# Is dose averaged LET really the best RBE descriptor for protons?

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Background:
For proton therapy, a relative biological effectiveness (RBE) of 1.1 has broadly been applied clinically. However, as unexpected toxicities have been observed by the end of the proton tracks, variable RBE models have been proposed. Typically, the dose averaged linear energy transfer ($LET_d$) has been used as an input variable for these models but the way the $LET_d$ was defined, calculated or determined was not consistent, which may impact the corresponding RBE value. This study compares consistently calculated $LET_d$ with other quantities as input variables for a phenomenological RBE model and attempts to determine which quantity can best predict the RBE value for protons.

Methods: Experimental set-ups of *in vitro* cell survival studies for proton RBE determination are simulated using the SHIELD-HIT12A Monte Carlo particle transport code. In addition to $LET_d$, several other beam quality quantities are determined, such as track averaged LET, $z^{*2}/\beta^2$, here called effective Q ($Q_{eff}$), Q and inverse of average kinetic energy. Each quantity was calculated using the dose as well as the track averaging methods. The type of particles included in the scoring of all quantities was also changed (i.e. all secondaries, all secondary protons or only primaries). A phenomenological linear-quadratic based RBE model is subsequently applied to the *in vitro* data with the various beam quality quantities used as input variables and the goodness of fit is determined and compared using a bootstrapping approach, for both linear and non-linear fits.

Results: Versions of $Q_{eff}$ and Q outperform LET with a statistically significant margin, with the best non-linear and linear fit having a relative root-mean-square-error (RMSE) for $RBE_{2\ Gy}$ ± one standard error of 1.77 ± 0.04 ($Q_{eff,\ t,\ primary}$) and 2.82 ± 0.08 ($Q_{eff,\ d,\ primary}$), respectively. For comparison, the corresponding best non-linear and linear fits for $LET_{d,\ protons}$ had a relative RMSE of 2.29 ± 0.06 and 3.41 ± 0.09, respectively. Applying Welch's t-test for comparing the calculated RMSE of $RBE_{2\ Gy}$ resulted in two-tailed p-values of <0.002 for all Q and $Q_{eff}$ quantities compared to $LET_{d,\ protons}$.

Conclusion: A more accurate RBE value for protons might be achieved by using versions of Q or $Q_{eff}$ as input for both linear and non-linear phenomenological RBE models.

# INFLUENCE OF DOSIMETER POSITION FOR THE ASSESSMENT OF EYE LENS DOSE

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**Keywords:**
eye lens monitoring, dosimeter position, thermoluminescent dosimeters

**Abstract:**
The objective of this study was to evaluatedifferent possible approaches of dosimeter wearing position in order to have a good estimate of the eye lens dose. Measurements were performedwith an X-ray system Siemens Nanodor and Hopwell Designs Inc. X82-160-MO-E, using a PMMA phantom to simulate the patient scattered radiation and a Randophantom to simulate the exposed worker. Thermoluminescence (TL) eye lens dosimeters were located on different positions of the Rando phantom to estimate the eye lens dose in typical work procedures. X-ray device and Rando phantom were positioned in a same way as during interventional radiology, with tube potential of 90 kV, current of 3 mA, 10 procedures in duration of 0,5 s. Dosimeters were placed on eye lens (with and without capsule), above and under canthus of eyes, left and right above eyes on the edge of medical cap, left and right on the thyroid collar, left, right and the middle of protective glasses. The results show that, for the studied conditions, best position for eye lens dose assessment is above left canthus of eye or position near left eye on protective glasses.

# Dose response of radio-photoluminescent glass detectors in 3 MeV and 5 MeV proton beam

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Our aim was to study the dose response of radio-photoluminescent (RPL) glass dosimeters in low-energy proton beams.

A 6 MV Tandem Van de Graaf accelerator connected to a vacuum chamber at the Ruđer Bošković Institute (RBI) was used to produce 3 MeV and 5 MeV proton beams with cross-sectional area 1x1 $mm^2$. These beams were directed at RPL glasses (type FD-7) with dimensions 16x16x1.5 $mm^3$ taken from environmental RPL dosimeters (type SC-1). The sensitivity of the detectors in the batch was checked before and after the experiments in the $^{137}Cs$ gamma field at the Secondary Standard Dosimetry Laboratory at the RBI. Standard annealing procedures were used and the dosimeters were preheated before dosimeter response values were measured with the FGD-202 reader.

Detectors were exposed to protons fluence ranged from $3.3x10^6$ to $3.3x10^7$ p/$mm^2$ and from $3.5x10^6$ to $4.5x10^7$ p/$mm^2$ for 3 MeV and 5 MeV, respectively. RPL signals were normalised to the number of protons irradiating the detector. The linearity index was calculated as a ratio of normalised RPL signal and reference normalized RPL signal where reference signal originates from the linear dose response range.

Linear dose response was confirmed for the whole investigated fluence range for 5 MeV and for the fluence range from $3.5x10^6$ to $2.2x10^7$ for 3 MeV. The corresponding dose linearity index was in the range of 0.95 – 1.05 whilst linearity slope is larger for 5 MeV compared to 3 MeV. For the fluence $4.5x10^7$ p/$mm^2$ of 3 MeV protons, linearity index of 0.59 was calculated suggesting sublinearity. Sensitivity of detectors was not changed after irradiations with proton beams and an average value ± standard deviation was 1.00 ± 0.02. The data collected in the linear dose response range was used to evaluate relative efficiency of RPL detectors to 3 and 5 MeV protons [1].

This work was supported by the Croatian Science Foundation (Project IP-2020-02-3593)

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# Evaluation of the neutron spectrum at flight altitude with Geant4 using different parameterizations

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In the last few decades, the development of aircraft with higher maximum cruising altitude and greater autonomy, as well as a significant increase of air traffic, has increased the problem of controlling the ionizing radiation dose level received by pilots, aircrew and aircraft electronics, and has begun to worry radioprotection and flight safety organizations [1]. The study of the effects of atmospheric cosmic radiation on avionics, aircrew and embedded systems detectors requires a detailed description of the radiation field incident on the aircraft. Using simulations based on the Monte Carlo code Geant4 [3], this work aimed to develop a virtual environment that allows the simulation of the transport of cosmic radiation and evaluate the influence of different parameterizations on the cosmic-ray-induced neutron on ground level and flight altitude. We have modeled the atmosphere from ground level up to 80 km, considering the effects of Earth's magnetic field. In order to do so, the atmosphere, the transport of the primary cosmic radiation (PCR) and secondary particles through the atmosphere subjected to the Earth's magnetic field were modeled. Methodologies were also developed to model the primary cosmic source incident in the atmosphere and to obtain the fluence rate and angular distribution of the cosmic-ray-induced particles as a function of altitude.

The results obtained from simulations were compared to experimental data from ground level and flight altitude, for different geographic regions and dates, evaluating the reliability of the physics model used to estimate the cosmic radiation transport in the atmosphere for energies above 20 MeV. Analyses were also performed of the influence of the Earth's magnetic field using the Geant4 code considering different versions and we also have considered different models of the atmosphere. As a product of this work, a virtual environment was developed that corresponds to a fraction of the terrestrial atmosphere up to 80 km altitude in a region of 50 km in diameter, which describes the fluence, composition, energy spectrum and angular distribution of the cosmic-ray-induced particles in the atmosphere as a function of altitude and space weather variables such as the solar potential, the local cutoff rigidity and the local intensity of the primary cosmic radiation incident. This platform models the primary cosmic radiation independently of external software, as the intensity of the PCR can be modulated using neutron counting from an on-ground neutron monitoring station, making the virtual platform standalone.

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# Bayesian statistics as a supporting tool in ionizing radiation dosimetry

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In order to reconstruct the dose absorbed in the body of a person exposed to ionizing radiation, physical dosimetry (including thermoluminescence dosimetry) and biological dosimetry are used. During emergency situations, environmental control measurements are often unenviable, e.g. because a detector is missing or contaminated, an unknown radioactive source or an unknown composition of a mixed radiation beam may appear. These are all examples of situations, where classical methods may be insufficient for a reliable exposure assessment. Therefore, there is a need to develop and implement statistical methods that enable a robust estimation of the absorbed dose. This study presents the utility of Bayesian statistics in a wide range of ionizing radiation dosimetry topics. It can be used to support classical techniques of exposure assessment, both in routine situations and in the case of a radiation emergency.

Bayesian statistics uses both a prior function, which can be assigned to the unknown parameter and a likelihood function obtained from an experiment, which together can be transformed into the posterior probability distribution of the sought parameter.

We present four examples of using Bayesian statistics in dosimetry. The first one concerns the methodology of dose estimation for mixed beta and gamma radiation, proposing an approach using thermoluminescence dosimetry and Bayesian analysis. It can be used in situations when the classical technique of dose calculation in personal dosimetry is insufficient.

In the next example, we estimate doses from mixed neutron and gamma radiation using cytogenetic biological dosimetry and Bayesian analysis, as well as Monte Carlo and the iterative method. It can be used in accidental or emergency situations of an exposed person in case of the absence of a personal dosimeter and physical measurement. Then the best technique to calculate the dose absorbed in the human body is the dicentric chromosome assay used in the current work.

The third method uses Bayesian statistics to assess the dose absorbed from a single type of radiation, which can be used in biological dosimetry in the absence of an appropriate dose-response calibration curve. In the methodology, the prior probability function is created from the few available curves from the literature and used to assess the dose.

The last method allows the use of the Bayesian approach for the selection of the most likely model describing the data as well as for curve fitting, for any data.

The tests we performed confirmed the methods based on Bayesian statistics work well, and they significantly improve the dose assessment, especially in the case of mixed radiation with unknown composition.

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# Development of a 14.8 MeV mono-energy neutron field in KRISS

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A standard irradiation field for 14.8 MeV neutrons is under development for mono-energy neutron standards research and detector calibration in Korea Research Institute of Standards and Science (KRISS), Republic of Korea. The 14.8 MeV neutrons are produced by T(d, n)$^4$He reaction (DT reaction) from a deuterium beam incident on a thin tritiated titanium (TiT) target [1]. In KRISS, a Cockcroft-Walton accelerator with a maximum energy of 400 keV will be installed shortly. In the meantime, we have developed a target chamber with beamlines and silicon surface-barrier detectors (SSD) for alpha particle measurements, called the associated particle (AP) detector system. The AP detector system and a Bonner sphere spectrometer will be used for neutron energy measurements.
We determined the design of the target chamber with the AP detector system using simulations and calculations. We performed DT reaction simulations on the target using TARGET software, developed by Physikalisch-Technische Bundesanstalt (PTB) [2], to estimate the emission rate and yield on various thicknesses of the target and injected deuteron energy. We determined the TiT target thickness and an operational deuteron beam energy to satisfy the aiming emission rate of neutrons on the target, ~ $10^8$ n/s. Furthermore, the beamline angle on the target chamber was optimized to minimize the influence of the angular straggling of the low-energy deuterons [3] by using minimum total folding angle, the angle between the neutron and alpha particle emitted from the target.
Based on simulations and calculation results, the target chamber with the AP detector system was manufactured. The performance of the SSD for alpha particle measurement was tested using an alpha source. The effect of a stopper film, which blocks inelastic deuteron particles to reach the SSD, was estimated. The alpha particles lose their energy on the stopper film, so the thickness of the film was optimized. We prepared a data acquisition system for alpha particles and neutrons. A new laboratory for mono-energy neutron standards was ready, as well. The ion beam accelerator is expected to be installed within a year. After the beam is installed and its stability is verified using a prototype system, the target chamber with the AP detector system will be installed and tested.
In this presentation, we will discuss simulation and measurement results on developing the 14.8 MeV mono-energy neutron field with the AP detector system in the KRISS. In addition, we will present the preparation status of the neutron field and measurement system and the laboratory.

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# Experimental validation of energy dependences of $YAlO_3$:Mn TL detectors: irradiation to ISO radiation qualities

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Various filters are commonly used to modify response of personal dosemeters, usually to flatten energy dependence or to cut unwanted energies or radiation types. However, in recent years new high Z materials possessing very strong energy response are studied as a candidate for energy selective detectors to be used, in particular, for emergency dosimetry. The effect of application of filters, made of different materials and various thickness is studied by Monte Carlo calculations using MCNP6.2 code, however, the calculate data need to be validated by experimental studies (benchmark tests). In view of potential type testing of new dosemeters utilizing high Z TL/OSL detectors, the measurements should be conducted with irradiations at standard photon sources in compliance with ISO 4037-1: 2019 specifications (ISO radiation qualities).

The poster presents experimental results obtained for $YAlO_3$:Mn high-Z TL detectors irradiated to different standard ISO radiation qualities (x-ray series of N-40, N-60, N-100, and isotope series of S-Cs and S-Co) modified by various metal (copper, aluminum, tin) filters of thickness from 2 to 6 mm. The experimental results are compared with results of Monte Carlo simulations done for the same 'radiation-attenuator-detector' combinations and geometry. Obtained results show good consistence between the experimental and calculated data that testifies adequacy of the used calculations and their applicability to modeling of modification of an output from the high-Z detectors exposed to photons of various energies.

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