IRPA 15

KARP The Korean Association for Radiation Protection

President's welcome

Program Schedule

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PL3

Health Risks and Their Uncertainties in Selected Scenarios of Exposure to Ionizing Radiation

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A recent UNSCEAR report analyzes uncertainties of health risks in scenarios of exposure to ionizing radiation (UNSCEAR 2020). The scenarios are derived from selected studies of leukemia incidence after CT scans of children in UK, leukemia and solid cancer mortality due to occupational exposure in the US, and thyroid cancer incidence in Ukraine after the Chernobyl accident. Selected life time risk calculations based on these studies of cancer are called ‚preferred risk inferences'.| Further, the report analyses circulatory disease mortality in Japan based on the life span study (LSS) of survivors of the atomic weapon explosions over Hiroshima and Nagasaki.

The preferred risk inferences agree well with calculations based on observations in the LSS. This is taken as confirmation of the transferability of LSS results to other exposure situations. The preferred risk inferences, however, have higher reliability and precision in the selected scenarios than the risk transfer from the LSS.

The report discusses sources of uncertainties such as quality of mortality and incidence data and various confounding factors. A semi-quantitative approach is used to estimate credible intervals of the lifetime risk estimates. The excess risk for leukemia in UK, for example, up to age 30 after four CT-scans in the age of one with a total bone marrow dose 20 mGy is assessed to 5×10^{-4} with a 95 % credible interval from zero to 20×10^{-4}. The radiation risk is comparable to the spontaneous risk of 9×10^{-4}.

In scenarios different from those of the preferred risk inferences, cancer risk estimates based on the selected studies and on the LSS agree less well. A main reason is the relatively short observation period in the selected studies. For scenarios with age ranges differing from those in the selected studies, the LSS keeps it outstanding relevance for risk inferences.

In the case of circulatory diseases, not much is known for several sources of uncertainties, and UNSCEAR did not attempt to estimate credible intervals of lifetime risk estimates. While an increasing number of epidemiological results have been reported on the risk of radiation-associated circulatory diseases in the past decade, the evidence on potential risk at doses well below 1 Gy is largely inconsistent and inconclusive.

Reference

UNSCEAR 2018 Report. Annex *Evaluations of Selected Health Effects and Inference of Risk Due to Radiation Exposure*. United Nations Scientific Committee of the Effects of Atomic Radiation. United Nations, New York, to be published (2020)

PL5

Update on the Fukushima Health Management Survey

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The Fukushima nuclear accident of 2011 released large quantities of radioactive materials into the environment. Many residents were forced to evacuate their homes and serious concerns regarding adverse effects on residents' health were raised.

The Radiation Medical Science Center at Fukushima Medical University has been commissioned by Fukushima Prefecture to conduct the Fukushima Health Management Survey, the purposes of which are to estimate radiation doses received by Fukushima residents and to monitor their physical and mental health conditions for the prevention or early detection and treatment of diseases and, further, to enable the long-term maintenance and promotion of the health of all residents in Fukushima.

The Survey is two-fold: one is the Basic Survey, which aims to estimate individual external radiation doses of Fukushima residents for the four-month period after the nuclear accident, and the other is the Detailed Survey, which is designed to assess the health conditions of residents and consists of the following four specialized surveys: 1) Thyroid Examination, 2) Comprehensive Health Checkup, 3) Mental Health and Lifestyle Survey, and 4) Pregnancy and Birth Survey.

In the Basic Survey, the external radiation dose received by each resident during the first four months after the accident is estimated based on information of where and when they stayed after the accident and the air dose rate at each location at the time of their stay. So far, dose estimates for more than 460,000 residents have been calculated, and it has been shown that 99.8% of the population received less than 5 mSv and 93.8% received less than 2 mSv. The highest dose received was 25mSv.

The Thyroid Examination covers approximately 382,000 Fukushima residents who were 18 or younger at the time of the accident. This examination is performed as a freely-selected examination under the explanation of the potential harm of the examination. Results of the first-round examination, which was implemented over the first three years after the accident and the second-round examination, which was implemented in the fourth and fifth years, have been finalized, revealing that 116 and 72 children were diagnosed with or suspected as having thyroid cancer in the first and second round examinations, respectively. The third-round examination is now reaching the final phase and 21 children have been diagnosed with or suspected as having thyroid cancer. The Prefectural Oversight Committee, which provides guidance and advice on the Fukushima Health Management Survey, has evaluated results of the first- and second-round examinations and concluded that no causal relationship can be established between radiation exposure and prevalence of thyroid cancer among Fukushima children. The Committee provided several reasons for this conclusion, which include low levels of radiation exposure, age distribution of patients, latency periods, and lack of dose-effect relationships.

There is, however, a global trend against thyroid cancer mass screening using ultrasonography for general populations because it is considered as having no benefit. Taking into account this trend as

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well as other ethical issues, the Oversight Committee is now discussing the future direction of thyroid cancer screening for Fukushima children.

Results of the Comprehensive Health Checkup have demonstrated an increase in overweight, high blood pressure, diabetes, and lipid abnormalities among evacuees, which suggests that evacuation itself has had significant impact on residents' health. The Radiation Medical Science Center for the Fukushima Health Management Survey, in cooperation with municipal governments, has been developing and implementing various health promotion programs in addition to implementing the Survey.

The Mental Health and Lifestyle Survey has demonstrated that the incidences of mood disorders, such as depression and trauma-related symptoms, and of children requiring special care due to their behavioral problems are higher in Fukushima than national averages. However, these percentages have been falling gradually with time and mental health support is being provided wherever necessary.

What has been made clear from the Pregnancy and Birth Survey is that the rates of preterm deliveries, low birth weight infants, and congenital anomalies in Fukushima are similar to national demographic trends.

We believe that it is necessary to continue the Fukushima Health Management Survey well into the future in order to understand the health effects of the nuclear accident and to develop, by utilizing these survey results, adequate health and medical countermeasures to maintain and promote Fukushima residents' health.

PL6

ICNIRP and Its Principles for Non-ionizing Radiation Protection

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ICNIRP recently published its updated principles for non-ionizing radiation protection (1). These aim at providing a consistent system of radiation protection over the entire radiation spectrum, taking into account the different types of health effects from different types of radiation. In this presentation, the principles for non-ionizing radiation protection will be presented and discussed.

Keywords: Non-ionizing radiation, Electromagnetic fields, UV radiation

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PL8

Future of the System of Radiological Protection: ICRP Perspective

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The system of radiological protection has evolved over many decades. It is based on the recommendations of ICRP, which consider scientific knowledge, ethical values and practical experience, as well as the work of many different organisations.

The current system has been effective and robust but faces new challenges with changes in modern society. The rapid development of the internet and social media means professionals, workers and the public increasingly expect instant access to accurate information. Keeping pace with this requirement takes significant resource, both human and financial, and this needs to be addressed if ICRP is to remain a key player. Producing relevant material on the website as well as increasing administrative skills is part of the process.

ICRP will continue to evaluate science and technical developments and will be working on different topics requiring guidance on radiological protection, such as veterinary medicine and space travel.

Ensuring young people are recruited into all fields of radiological protection is also important as ICRP needs experts across a range of associated specialties. ICRP has launched a mentorship programme, with mentees able to work with certain Task Groups, and it is hoped this will subsequently encourage nominations for Committee membership.

The last general recommendations of ICRP were published in 2007 (*Publication 103*), with work beginning a decade before. ICRP is now undertaking a major review, looking to refine the system of radiological protection in its next set of general recommendations. This is not intending to make significant change, but to examine all aspects of the system and to clarify and improve it where deemed necessary. The process will take several years and will involve considerable consultation, likely not just on the website but also through meetings and workshops. This will constitute the main work programme of ICRP for the next two terms and, hopefully, will strengthen the system, ensuring it remains fit for purpose until at least the mid 21st century.

SS2

ICRP and Women in Radiation

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ICRP is an organisation with over 90 years of history. The number of female members has gradually increased with time and now stands at 25%. However, women have had a very minor role in the more senior and leadership positions and it was 60 years before a woman joined the Main Commission. Six women have been Main Commission members, three have been a Committee Chair, and I am the first and only woman to be the ICRP Chair.

After 11 years in the role, I hope I have shown that women can have clear thought and vision to not only promote radiological protection but to also develop strategies and initiatives to improve the function of an organisation. This has been challenging and is only possible with the backing of a supportive team.

ICRP members are all volunteers and many women may be deterred from joining due to the extra commitment required, in addition to their daily jobs and family life. This should not be underestimated but many women, in all kinds of roles, manage complicated lives very successfully.

I am aiming to encourage women to join ICRP at a young age and one way is now through our mentorship programme and Task Groups, so they can learn how the organisation works and hopefully they will then consider nomination for ICRP Committee membership. Increasing openness and transparency will also show that women have much to offer and that ICRP is not just the domain of men.

SS2

Enhancing Radiation Protection Culture through a Gender Perspective

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The concept of culture involves ideas, habits, behaviours, knowledge, experiences, and attitudes, which are developed, shared and accepted by people in a society; and it includes both scientific and social dimensions. The main purpose of this paper is to raise awareness on the relevance of a gender perspective when understanding the development of every culture and, in this particular case, of the construction of a Radiation Protection (RP) culture. In this context, like most nuclear science-related disciplines, RP has historically been a male-dominated field and therefore, policies and strategies have been discussed, designed, and established primarily by a masculinized approach.

In spite of the general agreement on the relevance of diversity, women, while constituting over half of the world population, still remain underrepresented, especially in decision-making processes. In order to consolidate a RP culture, it is essential to integrate a gender perspective that ensures not only an active engagement of women, but also the identification and processing of their differentiated needs and visions, so that they constitute real, key components of this process.

In Radiation Protection, there is concrete evidence that women face obstacles mainly at four levels: socio-cultural, institutional, female subjectivity, and no gender solidarity. Firstly, there is a socio-cultural construction on the role women should play in a society that keeps them away from decision-making positions, and developing professional careers in Science, Technology, Engineering, and Mathematics (STEM) fields. This aspect is related to the female subjectivity, which is also socially constructed, and is part of the collective imaginary and of women's themselves who are raised to be mothers and carry out domestic tasks, according to the established stereotypes. Lack of self-esteem, fears, or insecurities are also elements of the education they receive from kindergarten. As of the institutional level, most regulations, statutes, organizational charts, are elaborated by and for men, so women naturally enter in a more hostile environment, where the lack of gender consciousness prevails. Last but not least, our society has historically promoted enmity among the diversity of women, so that they have not been able to develop strong tools for networking or teamwork among them.

In addition, applying a gender lens to RP is also crucial to cope with one of its most important challenges: public communication and empathy, which is essential for an effective implementation of RP measures. Women also remain having the more negative perception on nuclear, and especially on the effects of ionizing radiation. Thus, in order to bridging the gap of understanding between experts and the society as a whole, action plans need to include dedicated strategies, with its corresponding budget, for developing an innovative narrative and communicating from a gender approach.

In conclusion, it is vital to engage all key players, including high-level authorities, and educate them on the importance of gender mainstreaming in this particular field, considering that the above-mentioned obstacles have a direct negative impact not only on reaching gender parity, but also on the enhancement of a RP culture, and on achieving the required public acceptance.

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Enhancing Radiation Protection Culture through a Gender Perspective

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Keywords: Radiation Protection Culture, Gender Mainstreaming, Public Understanding.

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Special thanks to Ms Marina Di Giorgio who have really motivated and supported me to further study this topic by focusing my efforts on the particular characteristics and impacts on the RP field.

SS2

The Main Needs and Challenges Faced by Women Working in the Radiation Field, Raising Awareness of Possible Professional and Employment Opportunities

Renate Czarwinski
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The situation which women faces in working in radiation fields is embossed by the specific safety culture resp. radiation protection culture lived in the various resorts and facilities.
The statements in this short contribution for the Panel Discussion on Women in Radiation are intended to promote an interesting discussion on gaps, difficulties, cultural issues and successes which helps to overcome hesitations and old fashion behavioral patterns.
It has to be considered that women are an emerging economic strength in our modern world. What are there main needs for them? Firstly to mention is unrestrictedly the appropriate knowledge as well as the interest in the work/field. The woman who wants to work in radiation fields - independently from the technical, medical, regulatory or research resort - needs the adequate education and training for the job to know, to evaluate and to communicate the risks connected with the work. She also needs social and ethical competence and should avoid timidity. She must know what she wants and should have a vision how to implement it. Not to forget the financial resources e.g. in case of training for young professionals or for participation at important events like seminars or conferences.
Most of these needs are at the same time also challenges for women working in radiation areas – to mention is the issue of taking and balancing responsibilities and with no fear of failures. Also building of networks together with like-minded experts or participating in such networks e.g. to meet and justify important goals in radiation protection.
Raising awareness of possible professional and employment opportunities especially among young students in order to increase their interest, participation and commitment is a widespread field with responsibilities on different shoulders, e.g. the professor at the university should already have a commitment in radiation application and protection to wake up the interest of his students for this field. Methods/issues for raising awareness beside the internet job advertisements are inter alia

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The main needs and challenges faced by women working in the radiation field, raising awareness of possible professional and employment opportunities

Renate Czarwinski
President, German-Swiss Radiation Protection Association e.V.

- Networks
- RP societies
- Active participation in dialogues on selected topics
- Job fairs (employer, regulatory bodies, professional societies… should attend)
- Cooperation of professional societies with universities (holding lectureships,…)
- Internships, field trips

The criteria and the effectiveness of these topics will be discussed during the panel with the audience.

Special Session

SS2

Roles that Women Have Played in Radiation Protection from the IRPA YGN Leadership Committee and from China

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Radiation Protection is a comprehensive subject involving mathematics, physics, biology, chemistry, management and many other fields, in which women could make significant contributions. This work reviews the roles that women have played in radiation protection from the perspective of the IRPA Young Generation Network (YGN) leadership committee and from China. IRPA YGN leadership committee includes 3 women among 15 members, accounted for 20% of the ratio. A survey has been performed on the current percentage of the female members among the total YGN members in the IRPA YGN leadership committee member states. The preliminary results based on the data from 7 countries including Austria, China, Czech Republic, France, Japan, Korea and Spain show that women take up a considerable percentage, ranging from 15.4% by China to 67% by Spain. In China the Chinese Society of Radiation Protection includes 119 members of the council, and 12 among them are females. Some females are outstanding representatives of the institution or companies, serving as the leader of the radiation protection team. Chinese Nuclear Society holds Women Forum every 2 years as part of annual conference, and the forum titled 'Innovation and Dedication' in 2017 and 'Profiles of Women in Nuclear Industry of New Era' in 2019 respectively. On the whole, women have been playing a substantial role in radiation protection, and better promotion can be expected in the near future, if guidance in schools and public science popularization are undertaken.

Keywords: women, radiation protection, IRPA YGN, China

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SS2

Maternity Leave and Gender Equality-creating Opportunities for Women to Stay Connected with the Workplace Environment

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Not only in the nuclear field but in general, many women quit working after pregnancy, especially when there is a short maternity leave, which in some countries it is even unpaid, or they find difficulties in reconnecting with the working environment, when there is a long maternity leave. The latter sometimes leads to giving up working or looking for a new job.

Considering these facts, international scientific organization such as IRPA could help these women in overcoming the challenges which arise after returning to work. These organizations have great experience in long distance cooperation as their members are from all over the world and they work together taking advantage of all the available technology in telecommunications. The results of their work are outstanding and bring important improvements and recommendations in the radiation protection field.

Through this presentation I would like to emphasize some actions which could help women working in radiation protection overcome the challenges of returning back to work from maternity leave and help the employers and organizations such as IRPA benefit from their education, training and professional experience.

Keywords: maternity leave, women professionals

SS3

The Role of IRPA with the Development of Radiation Protection Culture for the Public

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Experiences from past emergencies, including the Fukushima Daiichi accident in 2011 and the following post-disaster recovery, highlight public understanding of radiation and risk as one of the most important challenges, and fostering and development of radiation protection (RP) culture for the public as well. The importance of enhancing our efforts on this challenges for the public has been recognised both by IRPA and the Associate Societies (AS). There is a growing need and interest for the AS to improve our ability to communicate effectively outside of our profession since the AS have a key role in supporting the RP professionals who are in the front line in the development of RP culture.

The IRPA AS (and RP professionals) are very well placed to play a lead role in the interface with the public since we are neutral – no advocacy (e.g. for nuclear). For example, in the challenging communications environment following the Fukushima Daiichi accident, the Japanese Health Physics Society immediately built a website named "Question and answers about radiation in daily life (Q&A)"[1)] to respond to every question on radiation and risk from the public and to answer all of them with scientifically accurate information from the viewpoint of RP experts. They received 1,870 questions in total, in which the top five frequent asked questions were about "exposure", "radioactive material", "radiation health effect", "child", and "decontamination". The RP professionals who were involved in the Q&A stayed in basic attitudes in responding to questions; using easy words (avoiding jargons) and feeling empathy with persons who questioned.

Another example is a development of "Information booklet for returnees"[2)], which was made by the group of experts and local authorities, whose expertise are RP, social psychology, agriculture, healthcare, and medicine, etc. This booklet introduced frequently asked questions from residents and their families who considered returning to their home and corresponding answers to them with practical advice and tips. Questions were about daily life such as "Can we open the window?"," Can we use furniture and tableware that we left home during evacuation?", etc. Answers were made based on scientific facts. First version was published in 2017 and the booklet was updated as to meet changing requests from residents with time in 2019. The booklet has been distributed to supporters (local government staff, counselors, etc.) who live and work in each local region and respond to requests from residents for consultation. RP culture has been fostered among returnees through this booklet.

It is important to ensure that these experiences are shared through the IRPA family so that good practices in the implementation of radiation protection culture and the communication of radiation and risk for the public are shared.

Keywords: *radiation protection (RP) culture, Q&A, Information booklet for returnees*

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SS4

ICRU Report 92 on Radiation Monitoring for Protection of the Public after Major Releases of Radionuclides to the Environment

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Nuclear installations may contain a large inventory of radioactive material, and an emergency may cause uncontrolled releases of radionuclides to the environment. Accidents are infrequent, but action needs to be taken quickly to mitigate the impact on public health, the environment, and commerce when a release occurs. Emergency preparedness requires that the operators of nuclear installations and the governments have radiation monitoring capabilities ready to use if an emergency occurs. The goal is to collect data quickly and accurately to inform decision-makers who must decide on protective actions.

A new report "*Radiation Monitoring for Protection of the Public After Major Releases of Radionuclides to the Environment*" has been prepared by the Report Committee 28 of the International Commission on Radiation Units and Measurements (ICRU) and published in 2019. The report provides detailed practical information on radiation monitoring to protect the people and the environment from harmful effects of ionizing radiation after significant releases of radioactive material to the environment. The report deals with the design and operation of off-site monitoring programs and systems. The report's material is based on the experience gained from responding to prior accidents that involved radioactive releases or potential radioactive releases combined with analyses of various policies and procedures used by countries worldwide. The experience accumulated after accidents at the Chernobyl Nuclear Power Plant (former USSR, 1986) and the Fukushima Daiichi Nuclear Power Station (Japan, 2011) is summarised. Accidents may also occur at other facilities or with devices containing significant amounts of radioactive material that result in large-scale environmental contamination and necessitate protective actions and radiation monitoring. The Goiânia accident with a medical source (Brazil, 1987) and accidents with nuclear weapons in Palomares (Spain, 1966) and Thule (Denmark, 1968) are also considered in the new ICRU report.

The discussion is focused on emergency and existing exposure situations caused by accidents at nuclear power plants, but the report may also be handy in case of other nuclear or radiological emergencies. Target users of the report are individuals and organizations responsible for the planning, design and operation of off-site radiation monitoring at national, regional and local levels. The report can also be useful for national authorities and organizations regulating and implementing the emergency preparedness and response and the environmental remediation of areas affected by an emergency.

Keywords: *nuclear, emergency, releases*

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SS5

Comparative Analysis of Medical and Biological Consequences of the 1957 Accident and the Techa River Contamination

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The present report is devoted to the comparative analysis of medical consequences of the 1957 accident and the Techa River contamination in the meaning of the effectiveness of the protective measures taken [1, 2]. Both accidents took place in the same Ural Region (Russia) at close time periods and were caused by atmospheric emissions (1957) and discharges of radioactive waste from Mayak PA into the Techa River (1949-1956). No emergency measures were taken on the Techa River: scheduled countermeasures were delayed and were not fully carried out. Scheduled activities aimed at ensuring safety precautions of the public after the Techa River contamination, included hydraulic engineering, administrative and agricultural measures [3].

Protective measures on the EURT were much larger than on the Techa River. Emergency measures on the EURT included evacuation of the population from the nearby settlements in the first days after the accident: rejection of food and fodder and their replacement with clean ones; determination of the density and boundaries of the contaminated territory; formation of sanitary protection zone, etc. Scheduled protective measures included additional relocation of the residents from the contaminated territories, decontamination of settlements and agricultural areas, development of radiation monitoring system and contamination control of food and fodder, reorganization and reorientation of local enterprises producing food products and others [2,4].

Interventions after the 1957 accident turned out to be significantly more effective and made it possible to significantly reduce exposure doses and minimize adverse health effects of the EURT population. For example, the content of ^{90}Sr, which was the main factor of internal exposure, was two orders of magnitude lower among the residents of the EURT than among the Techa River residents. The doses to RBM and stomach in the EURTC were significantly lower than those among the TRC members. However, it should be noted that not all countermeasures taken in the EURT were unequivocally effective. So, for example, resettlement of the EURT residents at a later time period (250-670 days after the accident) was ineffective in terms of the averted dose and could have an adverse effect on the quality of life of people. Resettlement 670 days after the accident prevented only 10% of effective dose and 20% of absorbed dose to RBM.

Early effects of exposure in the form of chronic radiation syndrome and tissue reactions (mainly from hematopoiesis, immunity, and central nervous system) were registered mostly in the Techa River residents [1, 3, 5]. The values of excess relative risk of malignancies and leukemias in the TRC and EURTC were also compared as the criteria of effectiveness of protective measures. It was noted that an increased radiation risk of malignant neoplasms and leukemias in accidentally-exposed residents of the Ural Region was caused by radioactive waste releases into the Techa River. Thus, the results of dosimetric, clinical and epidemiological studies of consequences of the two radiation accidents taken place in the Ural Region, Russia, show tremendous effectiveness of emergency measures after a radiation accident in case of their justification.

Key words: Accident, Radiation, Countermeasures

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SS5

Public Health Response and Medical Management of Internal Contamination with Radioactive Materials in Past Radiological or Nuclear Incidents – A Review

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Following a major radiological or nuclear incident, a large number of workers, responders or the general public may be exposed to external radiation and/or internal contamination with radioactive materials. Efficient and effective public health response and medical management are important aspects of controlling any subsequent impact affecting both short-term health outcomes and long-term health consequences from exposures. A systematic review of experience gained and lessons learned from the management of previous incidents can be used to investigate any gaps that need to be filled, in terms of knowledge and technology, and/or operation and implementation, to reduce the impact of any future incidents.

In this paper, public health response and medical management of internal contamination of 13 radiological or nuclear incidents were reviewed, based on the published literature and additional expert contributions, on topics covering emergency preparedness and response, including immediate protective actions, monitoring for internal contamination, assessment of radiation dose, medical management to mitigate health risks, risk communication, and ongoing health and medical follow-up. Each incident is objectively assessed to determine the aspects that were well managed and those that were challenging. The review followed by a critical examination on the reasons behind, taking account of the operational constraints at the time. The overall gaps, technological or operational, for managing internal contamination are summarized. The lack of evidence-based and internationally agreed protocols is identified. Recommendations on future efforts to fill such gaps are provided.

Keywords: Radiation emergencies, Internal contamination, Public health and medical management

SS5

Current progress and future challenges of Thyroid Ultrasound Examination Program in Fukushima: The Fukushima Health Management Survey

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The Great East Japan Earthquake on March 11, 2011 and its subsequent tsunami caused the accident at the Fukushima Daiichi Nuclear Power Plant, in which extensive damage to the nuclear power reactors resulted in massive radioactive contamination. The Fukushima Prefecture started the Thyroid Ultrasound Examination (TUE) program as a part of Fukushima Health Management Survey project in response to residents' anxieties for health risks by radiation exposure. The TUE program covers residents in Fukushima aged 0 to 18 years at the time of the nuclear accident.

This program consists of the primary examination and the secondary confirmatory examination. In the primary examination, screening of thyroid nodules and cysts was conducted with portable ultrasound apparatuses. In the second confirmatory examination, an interview about medical history, a physical examination, a detailed ultrasound examination, thyroid function tests, and urinalysis were performed. Fine needle aspiration biopsy (FNAC) was also carried out only when the sonographic findings of nodules or cysts meet the standardized guideline for implementation of FNAC.

In the Preliminary Baseline Survey (first round), 300,472 subjects were examined within 3 years after the accident and the participation rate was 81.7%. The proportions of the subjects who fell into the categories A1 (no nodules or cysts present), A2 (nodule ≤ 5 mm or cyst ≤ 20 mm diameter), B (nodule > 5 mm or cyst > 20 mm diameter) and C (immediate need for further investigation) were 51.5, 47.8, 0.8 and 0%, respectively; 2293 subjects in categories B and C were recommended to undergo the confirmatory examination. Of the 2091 subjects preformed the confirmatory examination, 116 were cytologically diagnosed as malignant or suspected malignancy. The prevalence of childhood thyroid cancer in Fukushima was determined to be 0.038% with no significant differences between evacuated and non-evacuated areas.

In the first Full-scaled Survey (second round) of the TUE, 270,540 subjects were examined until March 2016 and the participation rate was 71.0%. The proportions of the subjects who fell into the categories A1, A2, B, and C were 40.2, 59.0, 0.8 and 0%, respectively. Thyroid nodules cytologically diagnosed as malignant or suspected malignancy were found in 71 cases. There is no dose-response pattern between incidence of thyroid cancer and the geographical classification of estimated absorbed radiation dose in thyroid. Currently, we performed the secondary confirmatory examination of the second Full-scaled Survey (third round) and the third Full-scaled Survey (forth round).

We would like to mention on-going challenges in our program such as actions against a risk of overdiagnosis in thyroid sonography, psychological supports for participants and their guardians, and more sufficient explanation for shared decision making before examinations. With advices and suggestions from international experts, we intend to continue our efforts to fulfill our mission and consider how best the thyroid examination can serve Fukushima residents.

***Keywords**: Fukushima, Thyroid Ultrasoun Examination, Thyroid cancer*

Special Session

SS6 (T8.1-0040)

United States Regulations as They Apply To LASER Users, Strengths & Weakness, Lessons To Others

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In the United States laser regulations and standards play an important role in how lasers are used. If one wishes to sell a laser product in the United States, that product must conform to the U.S. Laser Product Performance Standard. Documentation of compliance must filed with the Center for Devices and Radiological Health, a branch of the U.S. Food and Drug Administration. Management of users of lasers in the work place need to provide a safe workplace. Oversight of this requirement is provided by the Federal or State Occupational Safety & Heath Administration. Which recognizes the American National Standard Institute Z136.1 Safe Use of Lasers as the consensus standard which compliance with demonstrates an adequate laser safety program. A number of additional laser guidance standards, specific to various laser applications also exists. These cover outdoor use, fiber optic communication, medical, light shows and research environments,etc.
While on the surface this all seems to blanket laser use and should be a satisfactory approach the reality is laser safety in the United States relies on the integrity of the manufacturer or use location.
As examples to support the statement above, the reporting of laser incident is fragmented and unreliable. OSHA's own database of laser workplace incident only contains 25 reports over a 40 year period and the majority are Nonbeam related. Laser manufacturers who submit the "certification" paperwork, can wait years before their submittal is reviewed. Once their submittal is sent to the CDRH, the manufacture can sell their product (since the submittal is only documentation the product meets the product requirements). One can be assured if a manufacture had to wait for CDRH review prior to selling a product the up-roar would have increased CDRH staff by a factor of 10.
Less than 10 of 50 States in the United States have a department or regulations on safe use of lasers within their borders. Most of these States just collect a registration fee and do little to no oversight. All this also extends to medical laser use. Biggest debate on the medical front is physicians objecting to beauticians being allowed to do laser based hair and tattoo removal.
Much of what is covered as US issues can be found in other countries. This presentation will shine light on the present status of laser safety oversight in the United States as well as making a number of suggestions to address the issue. As well as asking the question where is oversight needed and where can it be pulled back and how?

Keywords: Laser, Safety, Regulations

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SS6 (T8.1-0041)

ANSI Z136.8 Laser Safety For Reseachers, Recognizing an Unmet Safety Guideance Need

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In the United States several laser user standards exist. The major standard generating body is the American National Standards Institute. While the home organization is based in New York, it oversees the generation of standards under its umbrella name. The Z136 series is designed for laser user guidance.
The American National Standards Institute is a private non-profit organization that oversees the development of voluntary consensus standards for products, services, processes, systems, and personnel in the United States, established in 1918.
Outside the United States, the primary standard organization is the IEC. The 60825 series is the one most commonly cited for laser user guidance. While numerous laser user and recommended practices, fall under ANSI Z136 and IEC 60825 all but one, present their guidance as if all lasers are commercial products that meet government laser product safety requirements. That standard is ANSI Z136.8 Safe use of laser in research, development and testing.
Any user or Laser Safety Officer (LSO) is aware that laser used in the research environment, particular in development stage does not meet all the requirements of a certified laser product and many area controls are not always applicable or relevant. As an example, diode lasers on a breadboard (especially one that can be moved around) the idea of an emergency shut off switch is questionable. Especially in cases where the open beam path maybe a centimeter or less.
The ANSI Z136.8, recognizes the example above and gives the LSO support from the standard if they decide not to use such a device. Another good example is in the topic of training. The standard requires documented on the job training. Which any user will tell you it is on site training where safety takes place. Institutional laser training requirements are really only a first step of hazard awareness.
Last example is just a common-sense item. Both ANSI and IEC for access control of class 4 laser areas, support the use of room entryway interlocks. But neither require any operational check to see if they are functioning. If the entryway interlock is a safety device, don't you think there would be some requirement to test its functionality?
ANSI Z136.8 covers all these topics and much more. This presentation is to acquaint one with this standard and challenge users to have elements of the standard incorporated in other laser standards.

Keywords: Laser Safety, Standards, Research & Development, Laser Safety Officer

ACKNOWLEDGMENTS
ANSI-Z136.8 Committee members

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4. IEC 60825-1, Ed 2.0 (2014) Equipment classification and requirements
5. IEC 60825-14, (2014) Safety of laser products—users guide
6.OSHA Occupational Safety and Health Standards for General Industry, Code of Federal

SS8

Nuclear Disaster and Thyroid Cancer Screening in Fukushima: Lessons Learned and Proposal

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This report is a personal scientific view and does not represent Fukushima Medical University policy;
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When a nuclear disaster occurs, countermeasures are taken to reduce the health risk in the affected area. A health survey is also conducted to assess the degree of radiation effects and perform evidence-based interventions[1)]. However, those factors affecting carcinogenesis not only involve radiation but also environmental, life-style related, and socioeconomic factors. This complexity, including intervention, must be carefully considered for the health of the affected population. Post-disaster investigations and interventions have been increasing, even in situations other than nuclear disasters; however, these often cause disadvantageous for local residents. Thus the importance of developing codes of conduct for such investigations and interventions related to disaster public health has recently been emphasized[2)].

From 2011 to 2017, three rounds of thyroid examination were conducted among young people, who were screened by ultrasonography in the Fukushima Health Management Survey; more than 200 patients with thyroid cancer have been diagnosed. Our study showed that most cancers are considered to stop growing after an early growth phase[3)]. This large-scale mass screening resulted in innocent cancer diagnoses from among large growth-limiting cancer reservoir, even at young ages. Thyroid cancer screening is generally considered 'not recommended', by the US Prevention Services Task Force. According to WHO/IARC recommendation in 2018, thyroid cancer screening is not recommended to be unconditionally conducted even after a nuclear accident, and monitoring is desired for those who are estimated to be exposed to 100–500 mSv or more.

To prevent the harm of overdiagnosis, a change in the strategy from screening to voluntary monitoring is urgently needed based on a code of conduct, with better understanding of the natural history of thyroid cancer.

Keywords: *Overdiagnosis, Code of conduct*

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SS8

Features of Internal Exposure to Insoluble Particles Having High Specific Activity

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Micrometer-sized particles bearing radioactive cesium were found after the accident of TEPCO's Fukushima Daiichi Nuclear Power Station[1]. The particles are highly insoluble and have high specific activity as the activity per particle is ranged from a few to 100 Bq[1,2]. It is expected that the activity will move in the body as a single particulate material without dissolving to blood or tissue fluid when the particles are inhaled. Therefore, internal doses by inhalation of the particles depend on how the particles move in the body.

Internal dose estimation methodology of ICRP is designed for incorporation of countless radioactive nuclei. That is, numbers of disintegrations in source regions in the body are estimated by a deterministic way. However, the activity in the particles will not be distributed. Therefore, the ICRP's methodology cannot be applied to the inhalation of the particles when the number of inhaled particles is small.

The authors developed a methodology to estimate the numbers of disintegrations considering a stochastic movement of the particles named "stochastic biokinetic method (SB method)". This method enables to obtain a probability density function of internal doses by incorporation of the insoluble cesium-bearing particles[3]. This presentation reviews the outline of the SB method, and the topics and ongoing tasks regarding internal dose estimation for intake of insoluble particles.

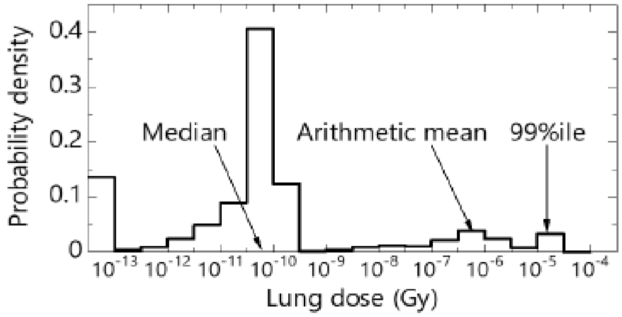

Fig. 1. Probability density function of lung doses for single particle inhalation.

Keywords: *Insoluble particle, Internal dose, Stochastic biokinetics*

ACKNOWLEDGMENTS

This presentation includes a part of results of a research supported by the fund of the Environmental Restoration and Conservation Agency (fund No.: 5-1501).

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SS9

Reanalysis of Cancer Mortality Risk in Association with Organ Absorbed Dose for Japanese Nuclear Workers 1991-2010

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Background: Japanese Epidemiological Study on Low-Dose Radiation Effects (J-EPISODE) has analyzed health effects in association with photon exposure assessed in $H_p(10)$ since 1990. However, it is under way to estimate cancer morbidity and mortality risk evaluated in organ absorbed dose, which is recommended by ICRP.

Aim: To reconstruct organ absorbed dose during 1957 to 2010, and reanalyze cancer mortality risk for J-EPISODE 1991-2010.

Materials and methods: The reconstruction method of organ dose principally followed the approach adopted in the IARC 15-Country Collaborative Study. However, the method was modified considering recent usage practice of dosimeters in Japan and body size of Japanese. Despite the IARC's framework with $H_p(10)$ being the common quantity, it was simplified using air kerma as common quantity (Figure 1).

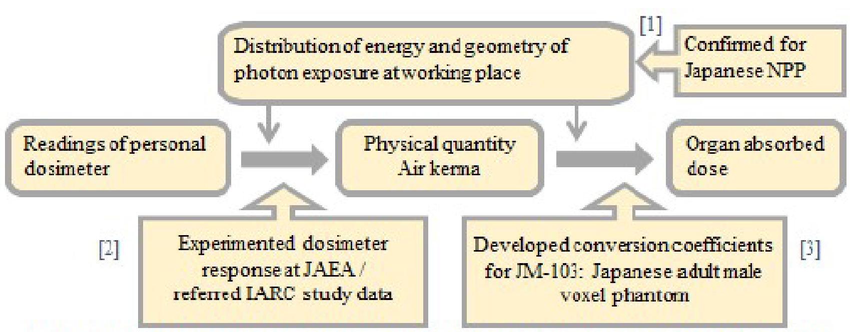

Fig. 1. Framework converting readings of personal dosimeter to organ absorbed dose

[1] The preceding studies on Japanese NPPs in 1980s were found to confirm that the assumptions of distribution of energy and geometry of photon exposure in IARC study were applicable for J-EPISODE.

[2] Dosimeter response data, defined as readings per air kerma, under combinations of a specific photon energy; 119, 207 and 662 keV, and a specific geometry; antero-posterior geometry and isotropic geometry, were newly experimented in the same way as IARC study for recently used three types of dosimeters; glass badge (GB), electronic personal dosimeter (EPD) and optically stimulated luminescence (OSL) dosimeter, while those for film badge (FB) and thermoluminescence dosimeter (TLD) referred IARC study data.

[3] Conversion coefficients from air kerma to organ absorbed dose were developed for Japanese adult male voxel phantom (JM-103) in order to compare with Caucasoid phantom.

[4] Finally, conversion coefficients from readings to organ absorbed dose were computed using the above data on dosimeter response as well as coefficients from kerma to organ absorbed dose for each year and each site where workers were exposed to photon, followed by reconstruction of organ absorbed dose for subjects of J-EPISODE during 1957 to 2010. Then, Poisson regression method was applied for estimating ERR (Excess Relative Risk) for cancer mortality.

Results: 1) The IARC assumptions of energy and geometry distribution were applicable.
2) Dosimeter response among dosimeter types demonstrated small differences.
3) Conversion coefficients for JM-103 revealed small differences from Caucasoid.
4) Conversion coefficients from readings to organ absorbed dose (Gy/Sv) were around 0.7 to 0.8.
5) Organ absorbed dose for several tissues was reconstructed from the recorded dose during 1957 to 2010.
6) ERRs for cancer mortality were estimated in terms of organ absorbed dose.

Conclusion: Evaluation method of cancer morbidity and mortality risk in association with organ absorbed dose, which is recommended by ICRP, became applicable for Japanese nuclear workers.

Keywords: Epidemiology, Organ absorbed dose, Nuclear worker

ACKNOWLEDGMENTS: This work was funded by Nuclear Regulation Authority, Japan.

SS11

SFRP-IRPA Workshops on the Reasonableness in the Practical Implementation of the ALARA Principle

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The practical implementation of the principle of optimisation of radiological protection (i.e., ALARA principle) was the subject of two workshops organised in Paris (France) in February 2017 and October 2018 at the initiative of SFRP within the framework of IRPA. The presentation summarizes the discussions and conclusions of these two workshops. The search for reasonableness was examined in three sectors: nuclear industry, medical practices and existing exposure situations. In all sectors, the optimisation remains a challenge and experience shows that this is implemented through a deliberative process to achieve a reasonable compromise with all informed parties. This issue was further investigated by three working groups – one for each sector – on the basis of cases studies. It emerges that, in complement to the use of classical tools such as cost-benefit analysis, the implementation of the optimisation principle implies a clear identification of the challenges to be met in order to achieve the best protection in the prevailing circumstances. These challenges may be specific to a type of exposure situation and in some cases to a given situation. The process should also well identify the relevant stakeholders and decision-makers to be involved and determine how they will be involved. A proactive process including development of awareness, empowerment and/or training may be needed. This reflexion deserves to be further developed.

Keywords: reasonable, radiological protection, optimization, ALARA

ACKNOWLEDGMENTS
The authors would like to express their acknowledgments to all participants in the two workshops in Paris in February 2017 and October 2018.

SS12

Radiation Protection in Suriname – Report on a Mission by the Dutch Society for Radiation Protection

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The International Radiation Protection Association (IRPA) encourages its associate societies (AS) to open membership to radiation protection professionals from countries where no AS is active. In 2019 the Dutch Society for Radiation Protection (NVS) took the initiative to determine the interest of RP professionals in the Republic of Suriname to participate in the activities of the NVS. This initiative coincided with the realisation of new legislation in Suriname on radiation protection along with the strong wish to develop an adequate training system for radiation workers and RPOs – lack of knowledge with respect to radiation safety matters is concerned a major problem in Suriname. Only recently, the Academic Hospital in Paramaribo has started a successful training program for their staff.

The NVS was invited by PAHO/WHO to visit Suriname for a mission with the aim to give recommendations on proposed legislation and on possible systems for education and training. The mission took place in November 2019 and ended with the first Suriname meeting of all relevant stakeholders. In this contribution, we will report on the outcomes of this mission and the potential IRPA-membership of radiation protection professionals in Suriname.

***Keywords:** IRPA, Education & Training*

SS12

Nepal & Other Countries without Regulatory Bodies on Radiation

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Nepal, one of the least developed countries with population of 26.6 million people is the biggest populated country among countries without a regulatory body. Radiations are widely used in medicine, agriculture, industry and other scientific research activities throughout the world. But in Nepal its use is almost confined to medical field for diagnostic and therapeutic purposes. In the absence of regulatory body, some hospitals face problems in purchase and transportation of radioactive material. News report on the disappearance of radioactive Iradium-192 disused source en route to Kathmandu for safety disposal to the country of origin, Belgium, has also raised a big question on safety & security of radioactive sources used for medical purposes in Nepal. All together fifty three countries, most of them least developed, still do not have any regulatory body. On 16th September 2015, Ministry of Education, Science & Technology, line ministry to International Atomic Energy Agency (IAEA) has issued one directives for regulation of radioactive materials in Nepal.

Nepal has yet to constitute rules and regulation as well as regulatory body in the field of radiation. But, Ministry of Education, Science & Technology (MoEST) has issued Nuclear Materials Regulatory Directive on 2015 to manage radiation sources. Under this Directive there is one recommendation committee to recommend MoEST to manage all nuclear related activities in the country. This happens to be the one and only document on regulation of radiation sources in Nepal.

Despite all the issues and the challenges inherent, the author remains confident over the eventual promulgation of the Nuclear Law and the formation of an effective regulatory agency. The government of Nepal has already tabled a final draft for Nuclear Law in the parliament for endorsement. Under that law, there is a provision for establishment of radiation regulatory body and hopefully Nepal will have regulatory body once the nuclear law becomes a law of the land.

The data shows that all together fifty-three countries around the world still do not have regulatory body or any nuclear law. Out of fifty-three countries twenty-four countries including Nepal are member states countries of IAEA and twenty-nine countries are still not member countries of IAEA. Data also shows that the highest number of countries including Nepal without regulatory body fall in Asia. The data shows that most of the underdeveloped countries still do not have any radiation regulatory body.

SS13

Chinese Standards and Risk Monitoring Quality Control on Radioactivity in Drinking Water and Food

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Food and drinking water safety monitoring are more and more concerned by public and government. Based on the food and drinking water monitoring for the health response in China following Fukushima Dai-ichi nuclear accident, started a routine risk monitoring and assessment of radioactivity in food and drinking water in 2012, from 8 provinces of NPPs vicinity to nationwide up to now. We organized the laboratories intercomparison once a year on gross α and gross β in water samples, prepared the food radionuclides reference materials (RM) for the measured results quality control.

National standard for drinking water quality includes action/screening level (gross α 0.5 Bq/L, gross β 1 Bq/L), which is on-going revised including uranium and radium 226. The national food safety standard includes reference level of natural and human-made 10 radionuclides, U and Th mass concentration, which is suggested to be revised, as well as the individual dose criterion (IDC) with health assessment for food and drinking water.

For radiation protection, it is necessary to strengthen the food and water radioactivity monitoring as baseline. This is a need to further standardized the emergency monitoring method, RM and uncertainty assessment for both emergency fast measurement and routine monitoring quality control.

Keywords: drinking water, food, control radioactivity

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SS14

Radon Dosimetry and Lung Cancer Risk Assessment: 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]. Although protection against radon is primarily based on measurement and optimisation, dose estimates are required for workers if, despite optimisation, radon levels in a workplace remain above the national reference level, or if, from the outset, radon exposure is considered as occupational 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 Publication 137 recommended a single rounded value of 3 mSv per mJ h m^{-3} (approximately 10 mSv WLM^{-1}) to be used in most circumstances, 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 the BEIR VI risk model 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, 3]. Given that the uncertainties from risk estimates are large, UNSCEAR concluded that its established dose conversion factor of 9 nSv per h Bq m^{-3} of equilibrium equivalent concentration of ^{222}Rn (1.6 mSv per mJ h m^{-3} or 5.7 mSv WLM^{-1}) should be retained for use in its comparisons of radiation exposures from different sources.

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.

Keywords: *radon, internal dosimetry, lung cancer*

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SS14

Thoron in the Environment - Methodology, Behavior and Dosimetry

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Radon isotopes include radon-222 (radon), radon-220 (thoron) and radon-219 (actinon). It is well known that radon-222 and radon-220 can be regarded as the main contributor of the annual effective dose for public in the world, which reaches 1.2 mSv and 0.1 mSv, respectively, as the worldwide average. However, the quantity of thoron is often observed more than radon in situations where thorium-rich building materials are unexpectedly used indoors. As the half-life of thoron is 55.4 sec, the health risk due to this isotope is considered to be negligible. In addition, there are no epidemiological studies on thoron exposures so far. On the contrary, radon epidemiology was well studied. Consequently, new scientific findings based on the latest epidemiological analyses for lung cancer risk due to radon exposure have been demonstrated. The residential radon concentration is mainly measured by passive radon monitors. Although such passive radon monitors are usually designed to detect radon efficiently and exclusively, several types of them can detect thoron together with radon. In this case, their readings may include both radon and thoron signals. If the readings are overestimated, the lung cancer risk will be given as a biased estimate when epidemiological studies are carried out. In this webinar, unique characteristics of thoron will be addressed.

TS-B1 (T1.B-0167)

The Health Physics Society's 'Ask-the-Expert' Feature: Widening Public Support through Empathy and Science

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A very large segment of the population is fearful of radiation, and sometimes rightly so. It's a word that conjures up images of something dangerous and invisible, and is often associated with the real fears of nuclear apocalypse that permeated the cold war era. TV dramas such as HBO's Chernobyl certainly fuel the fear response. Public response to radiological events ranging from true emergencies such as the Fukushima Daiichi nuclear power plant accident in 2011 to benign events such as the bucket of uranium ore discovered at the Grand Canyon Visitor's Center in early 2019 highlight the need for effective public communication strategies. All too frequently when an event is not considered dangerous by scientists, we fail to capitalize on the opportunity for public engagement. Public communication and empathy are some of the most important challenges that the health physics and radiation protection community face today. Empathy is of particular importance in effective public communication- understanding and explaining the science in layman's terms is insufficient to win public support. Rather, the ability to plainly explain the science must be coupled with an understanding of what the public or an individual is feeling about a particular issue. This requires more than science.

The Health Physics Society (HPS) sponsors a public information and outreach feature called "Ask-The-Expert" or ATE. ATE originated in 1999, and has grown exponentially since then to become the HPS's most successful public education endeavor, receiving over 1 million site visits annually. The origin of the feature and a discussion of the lessons learned during its first 20 years are detailed in a separate presentation. Here we focus on how ATE works, why it has been successful at building a culture of empathy, and how we are moving ATE into the future.

Ultimately, ATE works because of the tireless efforts of a small group of radiation protection professionals. These individuals have both solid technical prowess and an uncanny ability to distill complex science into understandable information. They take each question seriously, no matter how outlandish it may seem from a scientific point of view, and provide a personal response that starts with a simple bottom line answer to the question posed. Personal emails are written with the audience in mind – frequently someone has had some interaction with radiation and is scared, and the experts show compassion and provide invaluable knowledge on a personal basis. To reach the wider public audience, some questions and answers are posted and indexed on Google, allowing others to see them when searching for similar questions. The ATE webpages also contain nearly 60 fact sheets that cover everything from cell phones to space radiation exposure.

Moving forward, the ATE team, in collaboration with the HPS Public Information Committee, has begun creating a series of videos that cover the most commonly asked questions. Currently we have one video that covers diagnostic imaging and another that covers radiation exposure during pregnancy. Planned future video topics include radon and non-ionizing radiation, and this is likely to be a growth segment for outreach in line with modern media consumption practices.

The HPS's ATE feature is an effective method for widening public empathy and improving radiation knowledge. HPS is leveraging modern communication approaches in order to reach a broader audience and provide a trustworthy source for those searching an internet that is too often dominated by misinformation.

Keywords: Communication, public understanding, outreach

TS-B1 (T3.B-0590)

The Society for Radiological Protection (UK) Workstreams on Communicating Radiation Risk - Developing tools and guidance for the profession

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Across the Radiation Protection Profession and its allied fields, the communication of "Radiation Risk" is becoming an increasingly important skill. However achieving effective risk communication is becoming an increasingly challenging task given the somewhat negative public perception of radiation and conflicting views presented online and by both media and social media.

The Society for Radiological Protection (SRP) Annual Conference in May 2019 featured a workshop with a focus on "Communication of Radiation Risk in the Modern World". Three central scenarios were covered:

- Communicating to the Public Post a Nuclear / Radiological Incident
- Communicating Radiation Protection to Government / Local Authorities
- Communicating as part of Public Engagement Activities e.g. STEM

The workshop included technical talks and views from media specialists. The output of this workshop can be found in [1].

The workshop was well attended, and the feedback was promising with attendees requesting follow on workshops, and the development of subject specific guidance for Radiation Protection professionals.

IRPA is also currently showing an interest in this area and is developing a guidance document covering the guiding principles.

In summer 2019, SRP started a workstream aimed at developing a series of short specific user guides for the communication of radiation risk in certain scenarios, such as in support of Outreach, Emergency Preparedness or Medical Exposures.

The recent introduction of revised emergency planning legislation in the UK has the potential to result in the introduction of new or larger Detailed Emergency Planning Zones, and as such present an area of concern for the public, namely were they being protected adequately before, or is there a substantial increase in radiological risk? It was therefore decided that the first of the guides to be developed would be a "Guide to Communicating Radiation Risk in Emergency Preparedness".

The guide is to be developed via a workshop in November 2019 involving 15 attendees, including UK Government, Regulators, Media Specialists (including ex journalists, social media specialists, and specialists involved in communicating post actual incidents such at the Litvinenko poisoning), Nuclear Operators, UK Defense Operators, Radiation Transport Specialists and Local Authorities.

Following the workshop, SRP will pull together the draft "Guide to Communicating Radiation Risk in Emergency Preparedness" which will be sent to the wider participants for comment. Following resolution of any comments the guide will be formally published on the SRP website in PDF Format.

The proposed talk will provide an overview of the previous and ongoing work streams within SRP, and where the developed tools can be found for use by those working in the field of Radiation Protection.

Keywords: Risk Communication[1], Emergency Preparednes[2], Public Engagement[3]

ACKNOWLEDGMENTS

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TS-B1 (T7.B-0347)

Communication about Radon at Home: Main Pitfalls and Recommendations

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This presentation gives an overview of radon communication campaigns in Europe, identifies main pitfalls and concludes with recommendations for effective radon risk communication.

Empirical studies revealed that people living in high radon areas in Europe often find the risks of radon gas acceptable, despite the higher perceived risks: although they know that radon is bad for their health, they are not concerned about living in a house with high radon concentrations. When available, subsidies for home remediation are not fully used. Furthermore, campaigns aimed at increasing radon awareness showed that increased awareness does not automatically lead to action and behavioural change. For instance, after a series of communication campaigns, despite increased awareness, the concern about radon in their home has decreased. In addition, it has been found that, of those that test and find elevated radon concentrations in their home, only approx. 25% apply mitigation actions to reduce radon concentrations. Moreover, when radon measurements are recommended by authorities only small fractions of the population in the affected areas effectively carry out these measurements.

Implementation of National Radon Action Plans faces important challenges. While testing for radon and subsequent home remediation are scientifically and technically straightforward, empirical studies indicate that the levels of these actions are generally low. This indicates that radon remediation is not a scientific or technical problem, but rather a socio-political and psychological one, leading to a 'value-action gap'. A value-action gap refers to a situation where the values or attitudes of an individual do not correlate to his or her actions. The studies mentioned above provide evidence that changing public behaviour, in the sense of testing radon concentrations at one's home and applying mitigation actions such as the renovation of one's house, is challenging. The 'information provision approach', i.e. the assumption that individuals will act rationally in relation to the information provided, does not hold. Several health behaviour models, such as the Theory of Planned Behaviour, the Health Belief Model, the Protection-Motivation Model, and the Transtheoretical Model of Health Behavior Change, indeed suggest that effective radon risk communication should therefore seek to trigger other factors influencing behavior change in the target audience and not only to increase awareness about the radon issue.

The data presented were collected through quantitative (public opinion surveys) and qualitative studies (document review, interviews, round table discussions, workshops) in different European countries. The results show, that radon risk communication should focus not only on awareness but also on attitudes, subjective norms, self-efficacy and risk perception. Examples of communication materials are provided and recommendations for an efficient radon risk communication are presented.

Keywords: Radon, Risk communication, Communication campaigns, Health communication

TS-B2 (T2.B-0511)

Communication of Radiation Risk to the Public to Ensure its Realistic Perception

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Members of the general public, and even many radiation workers who are not appropriately trained in radiation protection, is usually confused in assessing the real risks associated with exposure to ionizing radiation mainly because the present system is too complicated to understand without some basic knowledge of atomic and nuclear physics. This is why laypersons are facing considerable difficulties to perceive the real risk associated with the use of radiation and nuclear technologies, where they are puzzled by mostly exaggerated information and news in tabloid press and some other means of mass communication following any incident, accident or misshape in the field. There are hardly found reports depicting the high level of the safety and security in using radiation and nuclear techniques in industry, medicine and many other areas where this technology is extremely beneficial. The paper discusses the progress in radiation protection achieved over the last few decades and compares it with the safety in other sectors where, in general, the level of protection of persons and the environment is in many cases at the lower level but this is usually not reported and presented to the public.

TS-B2 (T3.1-0093)

A Working Example Demonstrating the Benefit of Early Stakeholder Engagement

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There is an analogy that claims a camel is a horse designed by committee[1], the figure of speech hopes to emphasise the ineffectiveness of a limited committee of stakeholders that incorporates conflicting opinions into a single project. The distinguishing features of a camel and its poor temperament are the deformities that symbolise an operable project with underlying poor design. This workplace example is intended for Containment Engineers and Radiation Protection professionals, it cites an engineering committee commissioned to weld a considerable number of additional fixings on the inside surfaces of in-service high-risk radioactive process gloveboxes unfortunately early stakeholder engagement was not embraced resulting in the development of a camel type processes established in the absence of a Radiation Protection Adviser (RPA), a key contributing stakeholder.

The engineering committee opted for a stud-welding process which could be used near delicate in-service equipment noting the procedure had never been used on the inside of 'hot' functioning and grossly contaminated systems. Overseas engineers were contacted with only Rocky Flats[2] (RF) acknowledging they had carried out something similar, directing the committee to approve the RF solution. The RF solution was to introduce multiple welding sets and equipment into the containment system committing bulky equipment to ILW[3]. Where equipment was too large to travel through the connecting tunnel system, the committee's solution was to move the grossly contaminated equipment in and out of the line via glovebox widow removal or where this was not possible complete welding set were to be introduced into a target glovebox where it would remain trapped after use.

The committee belatedly engaged a Radiation Protection Adviser (RPA) to *retrospectively* approve their processes. It was clear they had not benefited from early RPA stakeholder engagement to advise against obvious radiation protection issues and so the project was frozen. A solution to ensure the powerpack remained outside the line was essential thereby requiring; only one which would remain free from contamination, limit ILW arisings and hazardous breaks of containment. The issue was how to electrically connect the external powerpack with the in-line welding gun. Normally the workstation's interface plate would be used but this was unavailable. The solution derived amongst the now RPA balanced stakeholder team together developed a Hypalon special glove port-cap with the required HT[4] feedthrough bonded into its centre. This straightforward breakthrough singlehandedly eradicated the use of multiple welding sets and resolved the need to use the interface plate, resulting in one welding powerpack being used on the outside of all gloveboxes. The special port-cap was specifically designed to replace a box-glove using existing changing techniques and equipment. The only item of equipment on the inside the active glovebox line was a small welding 'micro-pistol' reused as often as required with the only ILW arisings a box-glove and the special port-cap.

The benefit from albeit belated stakeholder engagement is obvious, undeniably earlier engagement would have more beneficial, highlighting critical analogies aimed at thoroughbred design committees should commend the benefit of early engagement and emphasise the selection of appropriate specialists; horse or camel? that depends on requirements.

Keywords: Stakeholder, Engagement, Glovebox.

ACKNOWLEDGEMENTS

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3. ILW – Intermediate level Waste contains higher amounts of radioactivity and in general requires shielding.
4. HT – High Tension – UK Collins Dictionary - Carrying, or capable of operating at a relatively high voltage.

TS-B2 (T3.8-0102)

Experience Gained in Translating Radiation Protection and Medical Physics Documents into Arabic Language

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Among the routine activities of Egyptian Atomic Energy Authority since its foundation in 1957 is training individuals in the field of peaceful uses of Atomic Energy. The main sources of training documents were from IAEA, ICRP ,ICRU, IOMP , EU Legislations UNSCEAR and IRR regulations and national laws and regulations. The need to translate some documents to Arabic Language is necessary to inform public and occupational worker as part of safety culture, training and education purposes such as teaching purposes in elementary, secondary schools and universities, as well as for postgraduate studies.

To simplify these scientific documents several books were written in Arabic. These books were widely used not only for training and teaching purposes but also as library reference materials. For more than fifty years more than ten books were published in Arabic Language not only from Egypt but also from other Arabic Countries. Furthermore basic radiation protection training courses were also written in Arabic by radiation protection Egyptian experts. With the cooperation between Arab Atomic Energy Authority, IAEA and Syrian Atomic Energy Commission, Basic Radiation Protection Training courses were conducted in Arabic Language and updated periodically.

In the present study attention is paid to review the experience gained in translating not only several ICRP documents and IRPA bulletins , Books and training courses written in Arabic Noting that the translated ICRP documents covered basically medical physics subjects and in progress the translation of 2007 ICRP recommendations of ICRP into Arabic Language.

TS-B2 (T3.B-0227)

The 2019 National Public Debate about Radioactive Waste Management in France: A Way to Involve the Public in a Sensitive Matter Mixing Technical, Political, Economic and Societal Issues

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Given the size of the French nuclear program, the management of radioactive waste is an important stake for France and a sensitive issue for the French public. It's also a very difficult matter to discuss due to the emotions and passion expressed by some of the stakeholders.

The French National Plan for Radioactive Materials and Waste Management (PNGMDR) is produced every three years, in accordance with a 2006 law. It constitutes an essential tool for overseeing and coordinating the management of radioactive materials and waste, ensuring a transparent, integrated and sustainable management of these substances regardless of their nature, their radioactivity level and their origin. It reviews the existing management modes, identifies the foreseeable need for storage and disposal facilities and determines the targets to be met for radioactive waste for which there is as yet no final management solution.

The PNGMDR is drawn up following discussions within a pluralistic working group which, under the joint chairmanship of ASN and the Ministry in charge of Ecology, brings together environmental protection associations, representatives of assessment and regulatory authorities and the producers and managers of radioactive waste.

According to new legal provisions, a national public debate was organized in France from April to September 2019 to discuss the 2020-2022 PNGMDR's contents. In France, such debates are organized by the National Commission for Public Debate (CNDP), an independent organization which produces a report at the end of the debate. This report summarizes the results of the debate to enlighten the PNGMDR authors. The CNDP is required to be neutral and shall not give its opinion during the debate.

23 public meetings were organized all around France. In parallel, the CNDP set up a focus group of 14 randomly selected citizens. These citizens were informed/trained by the CNDP about the PNGMDR's stakes during two week-ends and some of them attended the public meetings. The focus group produced a written statement which is included in the general conclusions of the CNDP.

The CNDP presented the first findings on the debate on September 25. The comprehensive report will be issued by the end of November 2019.

One of the conclusion is that it's very difficult in France to speak about nuclear issues. Some meetings were cancelled because of a too strong opposition.

A second conclusion is that only interested parties (nuclear opponents and pro-nuclear) participated. Regular citizens didn't participated, mainly because they consider that the radioactive waste issues are too complicated and because there's not a sufficient radiation protection culture among the French population. In this regard, there's a need of improvement.

One last conclusion is that there's a willingness over all the participants to the debate to be more involved in the preparation of the next National plan.

TS-B2 (T3.B-0399)

Radon in the State of Hesse – A Comprehensive Approach to Measurements, Public Information, and Protection

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In 2017 the Federal Republic of Germany adopted a new radiation protection act, the StrahlenSchutzGesetz[1] (StrSchG). This act incorporates a directive of the European Union (2013/59/EURATOM) into national law. It reorganizes the existing structure of radiation protection to reflect the current state of science and places stronger emphasis on natural occurring radioactive materials, especially, radon. Radon is the main contributor to the natural radiation background for the general population and number two health hazard for development of lung cancer.
Early on the federal state of Hesse in Germany developed a general comprehensive strategy to address radon protection. Different parts of which are required by the radiation protection act and additional parts to complement the former. The StrSchG calls for the the delineation of radon prone areas. To achieve this a large scope soil-gas measurement campaign[2] has been developed to determine the geogenic radon levels. This campaign is based on the underlying geological structure units and formations and is a collaboration with the geology department of the Hessian Agency for Nature Conservation, Environment and Geology (HLNUG). Furthermore, various scientific research projects will yield further understanding of the geological aspects of radon migration. Additional research on radon emission rates for indoor radon in small dwellings as well as radon migration in big buildings is being conducted.
The StrSchG requires that states initiate a communication strategy to inform the general public. After conducting extensive research into all the stake holders in the public discourse, as well, as the professionals involved in radon mitigation and education, the state of Hesse has decided on a comprehensive approach to meet this goal. The newly established radon center at the Technische Hochschule Mittelhessen (THM) will serve as a hub supervising the measurement campaigns, designing and carrying out the information campaigns and assist in the different research projects.
This presentation focuses on the administrative design process and the successful implementation of the radon action plan in the state of Hesse in Germany highlighting the combination of radiation protection, public education and scientific research. Detailed information will be given about the soil-gas measurement campaign as well as the implementation of the radon center.

Keywords: radon action plan, communication, center

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TS-B2 (T5.B-0609)

Scope and Researches from Viewpoint of Radiation Protection for STREAM-NST Education

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Nuclear science and technology (NST) is applied not only to energy supply but also to medical, industrial and agricultural fields. At present, the use of radiation and radioactive materials is indispensable for our daily lives. In order to support the diversity of radiation use, in addition to activities related to fostering public literacy, appropriate education on NST at secondary schools that leads to developing experts is extremely important. The horizontal development across regional and generational barriers and development of knowledge and technology in the field of radiation protection, which is one of the safety foundations has become an important issue for humanity. STREAM (Science, Technology, Robotics, Engineering, Art and Mathematics) is known as one of the keywords for actively promoting of secondary school education. In this research, each theme in the scope of radiation protection is arranged with the keyword "STREAM", and the latest research results are broken down from the viewpoint of secondary school education. We thought that presenting this information to stakeholders (teachers and students) appropriately and effectively increased their interest in the field of radiation protection and increased the overall attractiveness of this field. In parallel with the above, we are developing educational tools, modules and curriculum related to radiation protection that are applied at schools for secondary education, based on an international perspective in the Asia-Pacific region. Examples of typical developments in educational tools are: (1) Large-field Peltier-cooled cloud chamber, (2) Next-generation environmental radiation survey meter, (3) Simple radiation counter, and (4) Radiation sources made from natural materials. How can these tools and their respective educational modules be adapted and customized for each country and internationally spread to create the necessary environment for proper and safety handling in school education? We expect that this will be an important consideration for the future. For example, the appropriate use of natural radioactive sources in education is within the scope of risk management strategies for NORM, such as by the IAEA. Our oral presentation will introduce some of the progress in the development of these educational tools, present results of assessing their effects/impact based on implementational experiences in educational programs, and future issues/direction.

The researches were partially supported by JSPS Grants in Aid of Scientific Research (KAKENHI) in 2013-2015 (No. 25282034) and 2016-2018 (No. JP16H01813). Related researches continue, in several Asia-Pacific countries, by collaboration with international experts in the fields of radiation protection and secondary education from IAEA, USA, Australia, and Japan under the IAEA RAS/0/079TCP (2018-2021) with the support of NPO Science Technology Information Forum (Japan) and others.

***Keywords**: Radiation Education, Educational Tools, Secondary School Education, STREAM,*

TS-B2 (T6.B-0463)

Communication Challenges from Recent Radiological Events: The Cases of Ruthenium-106 and Iodine-131

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This presentation reviews how communication was managed in two recent radiological events in 2017: the detection of a radioactive cloud with level of ruthenium-106 (Ru-106) up to almost 1,000 times the normal amount and the detection of iodine-131 (I-131) at trace levels across Europe. The focus of the presentation is on how traditional and social media reported the event as well as the main lessons learned and recommendations.

In late September and October 2017, Ru-106 was detected throughout the northern hemisphere by national environmental radioactivity monitoring networks in several European countries and by the International Monitoring System established to verify compliance with the Comprehensive Nuclear-Test-Ban Treaty. Although low concentrations did not present any health risk to European citizens or harm to the environment, there was a public demand to "speculate" about the origin of the source and some criticism of lack of transparency and scepticism. Press releases were issued by some European nuclear safety authorities and the media reported the figures. There was a great mass media pressure on public relations officers in many European countries for around 40 days.

In January and February 2017, radioactive iodine was detected at trace levels in the air in Europe by the technicians of Europe's informal network of radioactivity surveillance experts. Some nuclear safety authorities in Europe and the European Commission reported measurements and declared that there was no potential threat to human health. However, the origin of the source could not be determined. Also in this case, there was intense media and public attention and increasing concerns over time despite statements from the regulatory authority that the level of radioactivity was negligible from the point of view of radiation protection.

The analysis of how communication was managed in these two radiological events of 2017 show that it is of paramount importance to communicate radiological events occurring beyond the national borders, even if they do not involve health or environmental impacts. This communication enables nuclear safety authority to lead communication in case of future rumours and/or excessive media coverage. The two cases resulted as an opportunity for national authorities to build trust in their expertise and competences.

For the ruthenium case, a documentary analysis was undertaken based on scientific articles, internet pages of European nuclear safety authorities, scientific communication by experts from different European research institutes, social media, media articles, official communication by national and international organisations and personal discussion with public relations officers in different countries. The information for the case of iodine-131 is mainly based on a review of internet pages of European nuclear safety authorities, research institutes, international organisations, on-line and social media.

Keywords: Risk communication, Radiological event, Ruthenium-106, Cesium-131

TS-B2 (T7.B-0063)

Community Oriented Risk Communication in Recovery Efforts after Radiological Contamination/Accidents

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The high levels of public worry and distrust of authorities, especially as associated with nuclear energy and radioactive waste, are well established in the literature and in the experience of risk communicators dealing with the general public. Though much of the experience with the heightened sense of worry that accompanies *planned* exposure situations associated with *proposed* radioactive/nuclear facilities (e.g. radioactive waste repositories, nuclear power plants, yet to be approved or built), there are also examples of *existing* exposure situations such as those following radiological accidents (e.g. Chernobyl or Fukushima). Some risk perception factors (e.g. trust, dread) are common to both *planned* and *existing* exposure situations, whereas others (e.g. volition, controllability) may be more prevalent in post-accident exposure situations. Specific risk perceptions held by a given population and its various subgroups must be acknowledged and incorporated into successful risk communication and public engagement strategies. Post-accident recovery programs and remediation projects, while in many countries requiring stakeholder and public acceptance to proceed, often fail to incorporate the specific risk perceptions held by the various affected groups in their risk communication and public involvement strategies. Such failures can foster a sceptical or angry public reaction, and hamper recovery in terms of both primary measures (radiological risk reduction interventions) and secondary measures (aiming at 'return to normal life', removal of stigmatization, health and well-being, etc.). A suggestion is made to deploy an iterative continuous improvement model for incorporating specific risk perceptions into risk communication programs in concert with mutual-gains based public engagement mechanisms.

Key words: Risk perceptions (psychometric paradigm), existing exposure situations, risk communication for the affected communities

TS-B2 (T7.B-0393)

Audience-centric Approach to Radiation Risk Communication Development, Case Radon Risk Communication

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STUK's strategic goal is that people understand the risks of radiation compared to each other and to other health risks. Our aim is to provide people with easy-to-understand radiation safety information to enable them to understand what is hazardous and what is not and consequently act correctly without unnecessary fear. Indoor radon is a significant heath hazard to Finnish population, and therefore radon communications plays a central role in STUKs communication activities to general public.

The communication environment has changed dramatically in recent years and people face a huge overload of information every day. In this environment, we can't make effective communication by shouting our messages a little bit louder. People search and use information they find relevant for themselves and listen to people they feel trustworthy. Successful risk communication requires ability to listen what is on top of people's mind and take part in the conversations, not just by replying but by understanding and creating contact with the target audience.

At STUK, we are looking at communication as a service to our target groups. It starts from developing a sense of empathy towards the people we are communicating to. In order to be heard, we need to connect our messages to the reality of our target audiences, and make it relevant for them, as well as utilize the channels and means of communication that our target audiences want to use. At STUK we use the service design methodology as a tool for communication development. It puts empathy and customer understanding at the centre of development and the process involves target groups in creating the solutions. This way, we can make more influential communication and make our messages easier to understand, remember and act on.

In Finland, the goal of radon communication is to increase understanding of radon risk and to strengthen correct behaviour. Service design approach was used to develop radon communication and during the last two years STUK has carried out innovative communication campaigns about radon safety and the efforts will continue. Design approach has also been used in overall development of risk-based inspection.

The presentation answers to following questions:

- How to build citizens' trust when communicating about radiation risks
- How to look at communications as a service to target groups
- How service design approach helps in creating empathy towards target groups
- How to develop risk communications in a holistic way, case radon at work places

Keywords: Service design, Risk communications, Radon communications

TS-B2 (T7.B-0451)

Risk Perception and Risk Acceptance: The Need of Effective Communications to Fill the Critical Gaps between Society and the Scientific Community

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Effective nuclear science communications can inform people about benefits and risks, allowing them to make decisions and choices. Nevertheless, regarding radiological protection, controversies and misinformation distort public's perceptions of radiation risks. This paper offers an overview of ineffective mass media communication and its impacts on citizens´ perceptions of nuclear technology. There seem to be a gap between society and the scientific community. The general public do not read high-specialized articles written by scientists. In Knowledge Society, where Internet and social media are the most common source of information, opinion makers seem to privilege sensationalistic rumors about the harmful effects of ionizing radiation, environment contamination and accidents. Moreover, available information about the biological effects of the exposure to ionizing radiation confuse public's perceptions of radiation risks and benefits. Whether experts agree that radiation causes observable health effects at high doses, Internet highlights controversies on the biological effects of low-dose radiation. Internet reports that the exposure to indoor radon is a risk factor for lung cancer. Internet reports that the radioactive monazitic sand brings health benefits. It is not easy for the general public to understand contradictions and to identify reliable sources. This article presents and discusses examples of pseudo-scientific information, newspapers errors, fake news, and anti-nuclear didactic material, where basic concepts are manipulated, and omission of vital information leads the public to mistrust and fear. Unfounded prejudices, misconceptions and misinformation are delivered in TV news, Internet articles, social media, TV series, cartoons and even through science journals. The general public, most often, do not have trustful information about radiological protection regulations and recommendations regarding human health, environment protection, management of radioactive waste or safe transport of radioactive material. Finally, this paper emphasizes the importance for scientists to be able to communicate to the public, developing science-based communication programs, evaluating the adequacy of those communications, investing in properly scientific divulgation about the risks and benefits of nuclear sciences that impact in citizens´ everyday life, such as medical applications, industrial applications, public safety and nuclear power generation. The balance between risk perception and risk acceptance depends on effective, trustworthy and understandable information. It is essential to educate educators and opinion-makers, combating fake pseudo-scientific information, social networks sensationalism and omissions of the media.

Keywords: Radiological protection, Risk perception, Public acceptance

TS-B2 (T7.B-0622)

The Making of "Information Booklet for Returnees" – Rebuilding Trust Through Collaboration with Local Communities

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Seven years have passed since the 2011 Fukushima Daiichi nuclear power plant accident. The nuclear accident caused irreversible damage to the life of local residents – their life was not only affected by the radioactive substances released from the nuclear power plant, but it was also thrown into confusion by the actions of authorities in the course of the nuclear disaster. Moreover, many faced the forced collective evacuation which changed the basis of their life completely. In this presentation, I first address social impacts of the prolonged evacuation on the construction and functions of local communities in Fukushima – particularly, the loss of traditional community, and trust in experts of science and medicine. Then, by re-defining their role in local communities as a mediator between experts and local community members, I discuss the crucial role of local public health workers in order to rebuild the broken trust between those two groups in the nuclear accident aftermath. And finally, I discuss our involvement in the process of trust (re)building through the development of "Information Booklet for Returnees".

Life after the nuclear accident – what involves collective evacuation, confrontation with radiation in everyday life and so on – is complex and diverse. To understand the dimensions of the problems that community members were facing and address their concerns properly, we needed to be directly involved in their everyday life and "grasp the native's point of view". The booklet was a mutual product of experts and people, and by extension, it became a means of communication between them in the nuclear accident aftermath. Therefore, the development process of the booklet – that was, our long-term active involvement in local communities – could be one of examples of (re)building trust between expert and people in the predicament of risk perception.

***Keywords**: Risk Perception, Risk Communication, Local Knowledge*

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TS-C (T3.C-0571)

The Role of Radiation Protection Societies in Tackling the Skills Shortage and Development of Young Professionals and Researchers

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Across the world we are seeing a resurgence in nuclear new build with a current estimated 160 new reactors planned for construction and an additional 300 proposed. In the UK alone plans are under way for the construction of 6 new commercial reactors, using 2 different reactor designs (EPR & HPR1000) and increasing investment in the development of Small Modular Reactors (SMRs).

This provides major opportunities for the current and future generation of Radiation Protection (RP) professionals. However, the scale of the nuclear new build ambitions, coupled with the high average age of the existing radiation protection professionals and increasing demand throughout the nuclear fuel cycle and wider radiation protection field (medical, radon, NORM, space travel) has heightened concerns of a skills gap [1].

The Society for Radiological Protection (SRP) has been undertaking an exercise to determine the scale of the skills gap within the UK. The results show that > 25% of the RP professionals in the UK are either retired or approaching retirement, with a further 50% retiring in the next 10 – 15 years. The International Radiation Protection Association's Young Generation Network has performed a similar analysis [2] with a similar concerning result.

Tackling the challenge of an aging profession is no simple task with the Professional RP Societies playing a crucial role in leading and coordinating efforts of RP Professionals and Employers. In order to increase the number of young professionals and researchers in RP will take time and requires 3 primary objectives to be met. These are summarised in Figure 1.

Since 2012, SRP has had an active outreach programme to encourage school students to become the future generation of RP professionals (Objective #1). This includes annual attendance at the UK Big Bang Science Fair with around 70 - 80,000 school children from all over the UK attending.

In addition to the outreach programme SRP has actively been working on initiatives to target Objective #2 and #3 including attending careers fairs, giving talks at various student events and conferences to help encourage them to see RP as a career path, and developing a mentoring programme for Young Professionals to provide careers advice and support continuous professional development. A number of our members are also involved in supporting the development of RP degree and apprentice programmes, and have established professional development programmes in their workplaces [3][4].

Despite this a far greater collective commitment is required to progress Objective #2 and #3 to increase visibility of the diverse nature of RP careers and future career progression.

Key to sustaining these activities is funding and resources (including volunteers). This highlights the importance of collaboration and coordination of efforts with other Professional Bodies, Governments, Trade Associations and Employers both in the UK and abroad for the benefit of the profession [1], in addition to finding alternative funding opportunities and avenues for public engagement.

TS-C (T3.C-0571)

The Role of Radiation Protection Societies in Tackling the Skills Shortage and Development of Young Professionals and Researchers

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The proposed talk will explore the scale of the skills shortage, provide an overview of SRP and its members' current activities to tackle this challenge, and a forward look at what additional initiatives are needed. We hope this will be of interest to attendees involved in educating and inspiring the next generation of workers and the public in radiation protection.

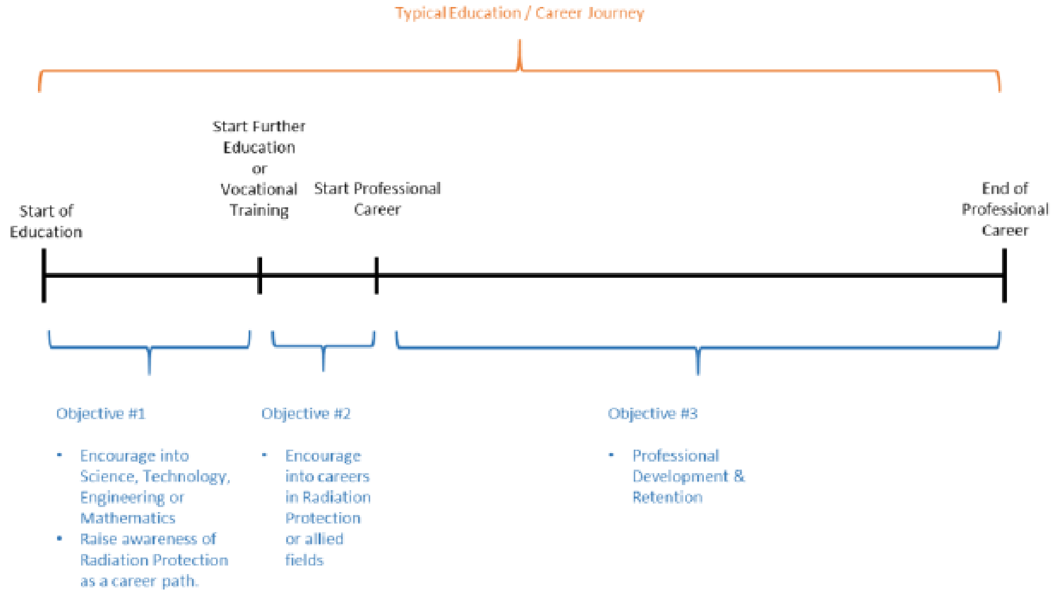

Fig. 1. Core objectives for encouraging and developing RP Professionals and Researchers

Keywords: *Human Capital, Young Professionals, Skills Shortage*

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ET1 (T2.2-0264)

Computational Human Phantoms and Micro/nano-Meter Scale Computational Dosimetry

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Computational human phantoms provide effective tool for computational dosimetry. In recent years, multiscale models including whole body phantoms, micrometer-scaled organ models and nanometer-scaled DNA models have been developed and applied in computational dosimetry. Organ dose can be obtained via Monte Carlo simulation with whole body phantoms. Monte Carlo simulation with the detailed micrometer-scaled organ model is able to provide not only the average dose of the organ, but also the detailed dose distribution inside the organ with complex structures. Moreover, the physico-chemical simulation using DNA model in nanometer-scale mechanically links the DNA damage with dosimetirc characteristics. Biological effects in heavy ion therapy are much more complex than photon therapy. RBE weighted dose represents the biological effect in heavy ion therapy. It can be estimated from different scales by combining macro-, micro- and nano-scale dosimetric simulations with multiscale models. In this study, a Carbon-ion therapy case is given as an example to estimate the biological effect using the multiscale computational dosimetric method. In this case, the dose distribution was calculated using the whole body phantom and the detailed organ model. The RBE weighted dose of each voxel in the whole body phantom and the detailed organ model was calculated based on the LET value of particles flying through the voxel. At the same time, particles flying through some voxels that representing different regions were recorded. These recorded particles were used as the source term for the microdosimetric spectra simulation and the nanometer-scale dosimetric simulation. RBE weighted dose in each chosen voxel was calculated based on the microdosimetric spectra. Nanometer-scaled dosimetric simulation was performed using NASIC, a nanodosimetry Monte Carlo simulation code. DNA damage spectra and chromosome aberration in the cell located in the chosen voxel were obtained to calculate RBE weighted dose using nanometer-scale dosimetric simulation. Results from different scale simulations could be compared to obtain a more accurate and comprehensive estimation of biological effect. This multiscale dosimetric simulation provides a method to elaborate the relationship between biological effect and physical dosimetric characteristics from multiple perspectives.

Keywords: multiscale models, multiscale dosimetry, heavy ion therapy

ACKNOWLEDGMENTS

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ET2 (T2.1-0211)

The Proposed ICRU Dose Quantities: EURADOS Impact Assessmemt

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The European Radiation Dosimetry Group, EURADOS (www.eurados.org) is a voluntary network of more than 70 European institutions and 600 scientists, that aims at promoting research and development and European cooperation in the field of the ionizing radiation dosimetry. As part of its strategic research agenda, EURADOS seeks to contribute to the development and understanding of fundamental dose concepts, such as the topic of operational quantities. Therefore, as part of its current work EURADOS is carrying out an assessment of the impact on radiation protection of the proposed changes to the operational quantities; we will also make recommendations on their implementation.

The proposed changes are being published in a joint report by the International Commissions on Radiation Units and Measurements (ICRU) and Radiological Protection (ICRP). These seek to address known problems with the existing operational quantities, for example regarding limitations in energy ranges and particle types, limitations which are becoming more important as the applications of ionizing radiation expand and evolve. The proposed operational quantities are conceptually different from the existing ones, being defined using the same anthropomorphic voxel phantoms as are used to derive the ICRP protection quantities. ICRU have carried out a consultation process and have revised the report in the light of comments received.

The paper summarizes the work done by EURADOS. In particular, the differences between the proposed and existing quantities are analyzed and their impact and application examined in the areas of: radiation protection practice, dosemeter and instrument design, calibration and reference fields, European and national regulation and current published standards.

Keywords:
Operational Quantities, ICRU, Impact

ET3 (T2.1-0390)

Lens of the Eye Dosimetry and Beta Radiation Protection Factors

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The Electric Power Research Institute (EPRI), Radiation Safety Program, Radiation Measurement and Dosimetry Research Focus Area has undertaken an evaluation of the issues surrounding dosimetry for the lens of the eye to support development of guidance and operational recommendations for radiological protection in nuclear power plant environments. The research is part of an ongoing program of work to inform and support radiological protection for the lens of the eye, in light of recommendations for reduced exposure limits from the International Commission on Radiological Protection and the U.S. National Council on Radiation Protection and Measurements.

The EPRI research examined several related questions, including the impact of various proposals for phantoms to be used in lens dose dosimetry, the response characteristics of several types of dosimeters available, and the angular dependence of dosimeters at various x and gamma ray energies and for beta particles. An agreed consensus has not yet been reached on the phantom to be used for dosimetry, but there seems to be some convergence to the use of a water filled cylinder 20 cm in diameter and 20 cm tall. Such a phantom was constructed, and compared with a solid Lexan phantom of identical dimensions to determine effects on backscatter and dosimeter angular dependence. With low energy x-ray radiation, a factor of approximately 7 percent was seen at normal incidence to the phantom, and 10 percent at oblique incidence, was seen between the water filled phantom and the solid phantom. The differences at typical gamma energies seen in nuclear power plants is much smaller, on the order of 1 percent. The results indicate that either solid or water-filled phantoms are likely appropriate for typical nuclear power plant gamma spectrum, and for beta radiation fields. Examination of the angular response at 0 degrees and 60 degrees showed a factor of about 2 for beta radiations, with a difference of a few percent for gamma energies. A variety of dosimeter technologies were tested, and while some designs seemed to perform better, all dosimeters can give acceptable results with proper calibration. Most all of the systems gave results within a factor of 2, even over a variety of energies and angles.

In nuclear power plant environments, the use of protective eyewear such as safety glasses, face shields, and respirators is typical. Protective equipment typically is constructed with several mm of plastic. While not useful for providing protection against gamma radiation, such materials afford the opportunity for protection from beta radiation. 2 mm of plastic was found to afford a protection factor of approximately 6, with higher protection factors seen with increasing thickness. The actual thickness of glasses and shields must be understood to know the degree of protection that is available. Further, protection presumes that plastic is between the source and the eye, placing importance on the use of side shields or wrap around safety glasses, and to the construction of face shields and respirators. A simplified approach to estimating protection factors was developed, and reported for use by utilities seeking to develop a robust and protective protection program.

Keywords: Lens of Eye Dosimetry, Angular Dependence, Beta Particle Protection

ET3 (T3.1-0434)

Findings from the 2019 IRPA Survey on the Implementation of the Eye Dose Limit

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In 2011 ICRP recommended a significant reduction of the occupational dose limit for the lens of the eye in planned exposure situations, from 150 mSv/year to 20 mSv/year averaged over five years with no single year exceeding 50 mSv. This recommendation has been incorporated into the IAEA BSS and the current Euratom BSS. In 2012 IRPA established a task group (TG) to identify key issues in the implementation of the new eye lens dose limit, and in 2015 a second TG was created to review progress in putting the recommendations from the early report into practice, and to collate the practitioner experiences. In 2017 IRPA published a guidance on implementation of eye monitoring and eye protection for workers, to provide practical recommendations about when and how eye lens dose should be monitored and guidance on use of protective devices related to exposure levels. IRPA agreed to continue to monitor and to ensure that the findings and issues highlighted through the TG can be an integral part of the ongoing international discussion. A third TG was created in 2018, and a new IRPA survey was launched in 2019 with the aim to collating results on actions taken by and position of the radiation protection community worldwide. A questionnaire was distributed, as a tool to structure the responses, addressing: *i)* the implications for monitoring and assessing lens dose and the interpretation of the results; *ii)* the implications related to the methods of protection being considered by different sectors such as medical, nuclear, and industrial applications, and the different personnel involved; *iii)* the direct or indirect impact on current practices, in relation to the implementation of the revised limit in the sectors of interest; and *iv)* the legislative processes being enacted or considered in relation to the limit for the lens, and guidelines or documents addressing eye lens monitoring. A total of twenty-five IRPA Associated Societies and ten national organizations/institutions, covering 44 countries in Africa, North and South America, Asia, Australasia and Europe, actively contributed by collecting, with their own internal procedures, views and comments from their experts and professionals, on the impacts related to the implementation of the new limit for the lens of the eye, and by filling in the questionnaire. The IRPA TG has analyzed the data to define and discuss common points, emerging issues and common concerns. The main results of the analysis will be presented and discussed together with an analysis of the trend and evolution of the views of radiation professionals.

Keywords: Survey on implementation of lens dose limit. View of professionals on implications in lens dose limit

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ET3 (T3.1-0527)

Practical Experiences of Regulators and Stakeholders Worldwide for Implementing the ICRP's Recommended Equivalent Dose Limit for the Lens of the Eye for Occupational Exposure

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Exposure to ionising radiation to the lens of the eye is a known risk factor for lens opacification, which may lead to a cataract. To prevent this effect, the International Committee on Radiological Protection (ICRP) has gradually revised the equivalent dose limits for the lens of the eye as more scientific data becomes available. Recently, epidemiological findings and animal studies have evidenced that tissue reactions for the lens of the eye have dose thresholds *ca.* 0.5 Gy that might be lower than previously considered. Therefore, for occupational exposures in planned situations, the ICRP recommended a decrease in the equivalent dose limit for the lens of the eye from 150 mSv per year to 20 mSv in a year, averaged over defined five-year periods (*i.e.* 100 mSv/5 years), with no single year exceeding 50 mSv (1). The main objective of the OECD Nuclear Energy Agency (NEA)'s Committee on Radiological Protection and Public Health (CRPPH) is to support member countries in identifying emerging issues, analysing their implications for radiological protection practices and regulation, and contributing to their resolution. The recommended revision to the dose limit for the lens of the eye by the ICRP in 2012 (1) have strongly impacted the radiological protection community, especially after the incorporation of the recommendation into both the international (2) and the Euratom Basic Safety Standards (3). In this context, the CRPPH decided to convene an Expert Group on the Dose Limit for the Lens of the Eye (EGDLE). The EGDLE commenced its program of work in July 2019, with the aim to provide an opportunity for regulators and stakeholders, *e.g.* nuclear and non-nuclear communities, to share lessons learned (both successes and challenges) in the practical implementation of the ICRP's recommended dose limit for the lens of the eye. The EGDLE will produce a report on these practical experiences. To assist the EGDLE to fulfill this task, a survey has been developed to facilitate the gathering of information from nuclear and non-nuclear regulatory bodies on the implementation of the ICRP's recommended equivalent dose limit for the lens of the eye for occupational exposures. The content and the first results of the survey will be presented.

Keywords: lens of the eye, dose limit, regulatory implementation

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We are grateful to all the members of the EGDLE for their fruitful implications in accomplishing the work: A. Rossini (Argentina), U. Oeh (Germany), D. Pollard (Ireland), M.A. Chevallier (France), S. Yokoyama (Japan), M.D. Rueda Guerrero (Spain), V. Rees (United Kingdom), J. Dillard (United States of America). M. Gomez-Fernandez, intern at the OECD-NEA Division of Radiological Protection and Human Aspects in Nuclear Safety, is also warmly thanked for his help.

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ET3 (T4.2-0289)

Assessment of the Occupational Doses to the Eye Lens in Interventional Radiology

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Objectives: To assess the annual lens dose received by interventional radiologists and to analyse the correlation between the lens dose and the doses recorded at the collar and chest level, as well as the correlation with the kerma-area product (P_{KA}).

Material and methods: During 18 months, three interventional radiologists worn 3 different types of dosimeters that were read on a monthly basis: 6 single-point optically stimulated luminescent (OSL) dosimeters calibrated for $H_p(0.07)$ [1] were placed both on the inside and outside of the lead glasses; an additional whole-body OSL dosimeter calibrated for $H_p(10)$ was worn over the thyroid shield; finally, 1 active solid-state dosimeter calibrated for $H_p(10)$ was worn on the chest over the lead apron. P_{KA} values were register on a local database both manually, in one of the rooms, and automatically, by means of an automatic dose management system, in the remaining two rooms.

Results: The main results obtained are summarized in Table 1. Radiologists' height and experience are given, together with the number of procedures performed during 2017 and the annual doses to the left eye, calculated both from the external glasses dosimeters (without glasses attenuation, -GA) and from the internal dosimeters (+GA). The slopes and determination coefficients (R^2) found in the correlation analysis are also included.

Table 1. Main results. In the correlation analysis, LD estimated from external dosimeters has been considered.

	Height (cm)	Experience (years)	Number of procedures 2017	Annual dose 2017 (mSv)		Correlation with collar dosimeter		Correlation with chest dosimeter		Correlation with P_{KA} ($\mu Sv/Gy{\cdot}cm^2$)	
				-GA	+GA	Slope	R^2	Slope	R^2	Slope	R^2
Radiologist 1	168	27	338	61±2	44±4	0.67±0.06	0.97	0.6±0.1	0.89	1.5±0.4	0.79
Radiologist 2	165	7	322	26±2	15±3	0.82±0.09	0.95	1.1±0.2	0.91	1.0±0.2	0.83
Radiologist 3	182	14	276	21±1	19±1	0.56±0.07	0.94	0.64±0.09	0.92	0.7±0.3	0.60

Conclusions: The 20 mSv/year limit [2] for the lens of the eye can be easily exceeded if no glasses are used. According to the determination coefficients, the collar dosimeter would be the best way to monitor the LD. However, important differences were observed among the slopes of the correlation analysis for different radiologists. For this reason, no common factor can be given to extrapolate the LD from the collar or chest dosimeters despite the high correlations found. However, on a local level, the LD of each radiologist can be estimate from the collar dosimeter's readings with high accuracy.

Keywords: *Occupational lens exposure, Interventional radiology, Paediatrics*

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ET3 (T5.3-0099)

Practical Implications of the Revised Dose Limit to the Lens of the Eye at Swedish Nuclear Power Plants

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In its Statement in April 2011, on Tissue Reactions [1], the International Commission on Radiological Protection (ICRP) issued new recommendations for a reduced dose limit for the lens of the eye in planned exposure situations. ICRP recommended, for occupational exposure in planned exposure situations, an equivalent dose limit for the lens of the eye of 20 mSv in a year, averaged over defined periods of five years, with no single year exceeding 50 mSv.

In the process of preparing IAEA´s TECDOC Implications for Occupational Radiation Protection of the New Dose Limit for Lens of the Eye [2] it became obvious that more information was needed concerning both the existing and potential eye lens doses at nuclear facilities and nuclear power plants.

The Swedish Radiation Safety Authority (SSM) requested, in 2013, Swedish nuclear facilities to survey, examine and ensure compliance with the reduced dose limit for the lens of the eye, on a national level. A system and a programme for monitoring the eye lens dose were to be in place, if and/or when needed. The aim of the examination was to ensure the correctness of the results when it came to dosimetric methodology, measurements and conclusions drawn. The aim was also to ensure that a plan for the future, regarding monitoring of the eye lens dose, in case the area and/or extent of the operations at the facility should change, exists as well. [3]

A survey in the form of a joint project was conducted by all Swedish nuclear facilities. Guidelines for when monitoring of eye lens dose is to be conducted were produced. As a result, a mandatory use of eye lens dosimeter was imposed for certain work categories.

A continued survey is still on-going and the results from Forsmark nuclear power plant, presented in this paper, show that for certain work categories, at boiling water reactors, the equivalent dose to the eye lens can exceed the effective whole body dose, measured by passive TL-dosimeter, by up to 50 %. Thus can dose to the lens of the eye be limiting when ensuring compliance with dose limits.

Keywords: revised dose limit for the lens of the eye, compliance with dose limits

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ET4 (T5.2-0296)

Determination of Nuclear Forensics Signatures from Lead Isotopic Ratios and Rare Earth Elements in Uranium Ore Samples using ICP-MS and LIBS

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In the nuclear forensic science context, signatures represent the characteristics of a specified sample of nuclear material that enables it to be distinguished or to identify its source. South Africa and Namibia are major exporters of uranium yellow cake. Hence, there is a need to develop nuclear forensic signatures in case of interdiction of nuclear material. Lead isotopic ratios and Rare earth elements (REE) have been shown to exhibit consistent patterns under varying geochemical and processing conditions and can therefore be used as signatures. The aim of this study was to determine nuclear forensics signatures from lead isotopic ratios and REE in uranium ore samples in South Africa and Namibia using inductively coupled plasma mass spectrometer (ICP-MS) and laser induced breakdown spectroscopy (LIBS). Soil and water samples were collected from uranium mines in Namibia and South Africa. South Africa lead isotopic ratios were found to fall within the national institute of standards and technology (NIST) standard while Namibia lead isotopic ratios were higher than the NIST standard. Namibia REE signatures exhibits light REE enrichment with fractionation while South Africa REE signatures exhibits heavy REE enrichment with fractionation. Critically developing these signatures can be used as a nuclear fingerprint for source attribution, basis of constructing a national nuclear forensic library and as a tool to resolve nuclear security cases.

Keywords: Nuclear forensics signatures, rare earth elements, nuclear security

ET4 (T6.3-0288)

Development of a Radiological Safety and Security Risk Index

Given the rising threat of radiological and nuclear terrorism, it is imperative to assess if radiological facilities, such as universities and medical centers, have the means to fully understand and evaluate the combined safety and security of their radioactive sources. In this context, risk assessment is a function of threat, vulnerability and consequences. This study aims to develop and demonstrate a methodology to compute a risk index for radiological facilities, based on the probability of occurrence of a Threat Event (TE) and its subsequent magnitude of incurred loss. This risk index provides a quantitative value for comparing risk and making decisions towards radiological safety and security improvements. The index employs inputs that include a set of threats, vulnerabilities, and consequences. These were used to construct a single composite number by weighing the threat scenario probabilities, relative attractiveness and characteristics of the radioactive material, multiple parameters elevating vulnerability of source security, and the consequence net loss. The risk decomposition is based on the Factor Analysis of Information Risk (FAIR) ontology. Probability density functions and event trees were then used to simulate scenarios to estimate the probability of successfully completing a malicious act at the university, such as theft of the source. To demonstrate this index, a higher education institution that uses a number of radioactive materials for research and teaching, was analyzed. Specifically, three facilities housing nuclear or radioactive sources at the university were compared: a research reactor facility, Co-60 irradiator, and radiopharmaceutical laboratory. The emphasis of the study is on the research reactor, but the other facilities were also analyzed for comparison. The research reactor facility houses a 10-kW swimming pool type reactor containing plate type uranium/aluminum fuel. The facility also houses other fuel and radioactive sources needed for operations and research. The irradiator facility contains both Co-60 and Cs-137 sources with TBq amounts of activity. The radiopharmaceutical facility contains a number sealed and unsealed sources with GBq amounts of activity. Two proposed safety (equipment malfunction and human error accidents) and security (malicious attack of theft and sabotage) scenarios were simulated for each facility. The radiopharmaceutical laboratory sources yielded the highest probability of both successful sabotage and theft outcomes as well as probability of accident. The reactor facility yielded the highest consequences in the sabotage scenario. The contribution of the proposed research is significant as it allows for a new tool in the field of coupled radiological source safety and security-one that is expected to introduce, analyze and numerically test a methodology that yields a facility level risk index.

ET5 (T3.1-0268)

IRPA Task Group on Radiation Protection in Industries Impacted by NORM

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IRPA is the international voice of the radiation protection profession with the aim of promoting the worldwide enhancement of professional competence, radiation protection culture, and practice. This is achieved by providing benchmarks of good practice and encouraging the application of the highest standards of professional conduct, skills, and knowledge for the benefit of individuals and society.

An emerging and growing concern in the field of radiation protection is naturally occurring radioactive materials (NORM).

Early in 2019, IRPA established a Task Group of internationally recognized radiation protection practitioners working with operational, regulatory and policy aspects of NORM.

The working group aims to increase awareness of issues related NORM in industry and to develop a common understanding of requirements, good practice and challenges faced by industry practitioners and regulatory bodies. This is all within an international context where significant diversity of national and technical conditions exists. A key objective is to deliver regulators and practitioners appropriate information to ensure that controls for NORM are commensurate with the actual radiological risk.

The working group works to;

- Identify problems and difficulties in the industry and trade resulting from natural radioactivity.
- Review and promote the existing library of good practice documents for radiation protection in various industries impacted by NORM, and where appropriate to develop additional guidance.
- Consider the practical gaps related to ethical foundations of radiation protection in working with NORM and stimulate open discussion of such issues, and to promote practical solutions which apply the graded approach to regulation.
- Engage with industry and collaborate and co-ordinate with other groups of similar interests in this field, including IAEA.

Since commencement, the task group has completed a number of tasks and this paper provides an update on the activities of the working group.

Keywords: NORM, Task Group

ET5 (T3.7-0117)

The ICRP Approach for Protection from NORM Exposure

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Abstract: The International Commission on Radiological Protection (ICRP) recently approved a new Publication providing guidance on radiological protection in industries involving NORM (future Publication 142). These industries may give rise to multiple hazards and the radiological hazard is not necessarily dominant. The industries are diverse and may involve exposure to people and the environment where protective actions need to be considered. In some cases, there is a potential for significant routine exposure to workers and members of the public if suitable control measures are not considered. Releases of large volumes of NORM may also result in detrimental effects on the environment from radiological and non-radiological constituents. However, NORM industries present no real prospect of a radiological emergency leading to tissue reactions or immediate danger for life. Radiological protection in industries involving NORM can be appropriately addressed on the basis of the principles of justification of the actions taken and optimisation of the protection using reference levels. An integrated and graded approach is recommended for the protection of workers, the public and the environment, where consideration of non-radiological hazards is integrated with the radiological hazards, and the approach to protection is optimised (graded) so that the use of various radiological protection programme elements is consistent with the hazards while not imposing unnecessary burdens. For workers the approach starts with the characterisation of the exposure situation, and the integration, as necessary, of specific radiological protective actions to complement the protection strategy already in place or planned to manage other workplace hazards. According to the characteristics of the exposure situation and the magnitude of the hazards, a relevant reference level should be selected and appropriate collective or individual protective actions taken. Exposure to radon is also treated using a graded approach based first on application of typical radon prevention and mitigation techniques, as described in ICRP *Publication 126*. A similar approach should be implemented for public exposure through the control of discharge, waste and residue, after characterisation of the situation. If the protection of non-human species is warranted, it should be dealt with after an assessment of radiological exposure appropriate for the circumstances, taking into account all hazards and impacts. This should include identification of exposed organisms in the environment and using relevant derived consideration reference levels (DCRL), to inform decisions on options for control of exposure.

ET5 (T3.8-0251)

The Effectiveness of Radiation Safety Culture in the Prevention against the Occupational, Public and Environmental Effects of NORM Industries

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Concerning the majority of human activities handling raw materials and minerals containing an amount of radioactivity, the level of exposure due to primordial radionuclide and their decay series has not considered as a radiation protection concern. However in some situations an enhanced radiation exposure may occur as a result of the extraction and the industrial processing of the raw materials, in this case these materials are used to be referred as NORM, the acronym of "Naturally Occurring Radioactive Material". The enhanced radiation exposure of NORM has the potential to cause an occupational, public and environmental harm, consequently an international consensus in several levels on NORM exposure management is needed. In this work we will highlight the effectiveness of the radiation safety culture in the industrial activities involving NORM in Morocco as a case study. For that we will answer the following questions: What are the important points on the reinforcement of the radiation safety culture in NORM industries? Who is the responsible? How we can implement a strong radiation safety culture and successful training between the workers?

Introduction:

The level of exposure due to primordial radionuclide (Uranium- 238, Thorium- 232) and their decay products containing in raw materials, feedstock and minerals has not considered as a radiation protection concern. However in some cases, an enhanced radiation exposure may occur as a result of the extraction and the industrial processing of the raw materials. In this case, these materials are used to be referred as NORM the acronym of Naturally Occurring Radioactive Materials.

Purpose:

The primary objective of this study, is to reinforce the awareness about the important role of the radiation safety culture as one of the prevention keys against the radiation exposure in NORM industries

Methods:

We study the Phosphate Industry as a NORM industry in Morocco as a study case; which we implement a radiation safety culture and training program/model for the case of Phosphate industry taking into account the properties and characteristics of the industrial process. This approach should be a graded response starting with the characteristics of the current situation; the strategies already existed in radiation safety culture point of view. For that we answer the following questions: What are the important points to reinforce the radiation safety culture in NORM industries? Who are the stakeholders responsible for the radiation protection in NORM industrial processes? What is their role so? How we could implement a radiation safety culture program and efficient training into the industry?

ET5 (T5.1-0456)

Naturally Occurring Radioactive Material: The Challenge for the Current Radiation Protection System

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Rio Tinto

Radiation is ubiquitous in nature and driven by the uranium, thorium and potassium series. These radionuclides can be enhanced my natural or man-made processes and give rise to Naturally Occurring Radioactive Materials (NORM). The current radiation protection system was designed to protect against exposure from man-made radiation generators and artificial radionuclides. Radiation protection from NORM has had to fit into this system and the fit is not perfect. NORM is generally associated with large quantities of low specific activity material and this gives rise to different management approaches across its entire use cycle: natural state, mining, processing, transport, use, re-use. And disposal. NORM is also associated with a wide range of industries not generally considered as associated with radiation or the nuclear fuel cycle. In most of these industries, the risks associated with radiation are minor in comparison with more traditional safety and occupational health hazards. Even the simplest of decisions around whether it is an existing or planned activity is fraught with difficulties. A "one size fits all" approach, consistent with the current system of radiation protection, is not appropriate for a large number of industries. Processes, such as the graded approach, may become an integral part of ensuring safety and radiation protection from NORM sources. Radiation protection, associated with NORM, needs a combination of awareness and a practical implementation approach which cuts across international organisations, regulators, practitioners and other concerned groups.

ET5 (T5.6-0286)

Cradle to Grave of Regulatory Oversight on Decommissioning of Facilities Handling NORM in South Africa

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South Africa has a long gold mining history, the precious mineral contributes to the most important part of the country's economy. However, the extraction of gold has excavated important volume of uranium and its daughter products. For decades, the mining industry prospered without a radiological regulated framework resulting in widespread contamination. The South African National Nuclear Regulator (NNR) was established in 1999 to provide for the protection of persons, property and the environment from the harmful effects arising from ionizing radiation produced by radioactive materials. The NNR is responsible for granting authorisations and exercising regulatory control on Naturally Occurring Radioactive Materials (NORM) activities such as mining and processing of radioactive ores. The NNR has developed regulations and guidance, on safety standards and regulatory practices to provide requirements and guidelines based on the IAEA Safety Standards to be applied by authorisation holders when undertaking decommissioning and rehabilitation or remediation. A formal process was established comprising: site characterization, public and worker safety assessments, criteria, strategies (choice of technology), surveillance and waste management programme including requirements for, funding etc. The Regulatory control is exercised from the beginning of the process to the end using regulatory reference levels for surface contamination and specific activity. Activity limits for unconditional release are established at levels of radiological risk which are below regulatory concern and do not warrant further regulatory control. However, owing to the nature of mining and mineral processing activities in general, practical and economic realities may not enable the complete achievement of the decommissioning and rehabilitation objectives, leading to derelict and ownerless mines, and unrehabilitated areas. These legacy sites present a radiological safety and security hazards aggravated by the illegal mining activities, for millions of people as no control and protection are applied. The South African Government took action with the establishment of cooperative governance agreements between the Regulator and relevant Organs of State to impose appropriate controls for new and operating facilities to avoid creation of new legacy issues. Although, the implementation of the regulatory process by NORM facility authorization holders led to a decrease in activity levels and a possible grant of a release certificate, the country has embarked on a long journey: Implementation and enforcement at a national level is still challenging as 5700 derelict and ownerless mines in South Africa may require 800 years for remediation/rehabilitation at an estimated cost of R100 billion.

Keywords: Decommissioning, NORM, Remediation/Rehabilitation

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ET5 (T5.7-0697)

Proposal for NORM Treatment and Final Disposal in Brazil

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In Brazil, the normative responsibility on the NORM subject belongs to CNEN - National Commission of Nuclear Energy. In this country, no practice to address this kind of waste is currently allowed, so NORM wastes are stored on the platforms/FPSOs, causing enormous damage to the petroleum industry. Therefore, in view of this technological impasse, the viability of an innovative treatment that meets the environmental, social and commercial needs is assessed, solving, definitively and safely, the final disposal for this kind of waste. The developed technology can be divided into five subprocesses: 1- Radiochemical analysis to determine the activity concentration of radionuclides in the crude sample; 2- Drums containing the waste go to the area of sieving and packaging in trays, which will go to the static oven; 3- Thermal desorption of the organic material using the correct temperature; 4 - Gases are sent to a post-combustion chamber to be destroyed; 5 - The remaining material containing the radionuclides is prepared for dilution with inert material until the concentration reaches the permitted activity concentration limit for disposal of radioactive waste in industrial landfill. For radiochemical analysis, eight samples of oily sludge from two different origins were obtained, and the activity concentrations of Ra-226 and Ra-228 were determined by gamma spectrometry, before and after the treatment. The results are showed in Table 1 and Figure 1 below. These results suggest that the radionuclides are not eliminated with the organic matter, being concentrated in the inorganic one, allowing the mentioned procedure of dilution to be performed.

Table 1. Results – activity concentration for Ra-226 and Ra-228 in the analyzed samples.

Samples	N		Radionuclides (kBq kg^{-1})	
			226 Ra	228 Ra
ALPAR 104 (treated)	2	Mean ± SD	4.804 ± 0.156	2.659 ± 0.94
ALPAR 105 (crude)	2	Mean ± SD	3.739 ± 0.122	2.036 ± 0.074
LB-983 (crude)	2	Mean ± SD	0.135 ± 0.0006	0.159 ± 0.001
LB-984 (treated)	2	Mean ± SD	0.209 ± 0.0009	0.249 ± 0.0009

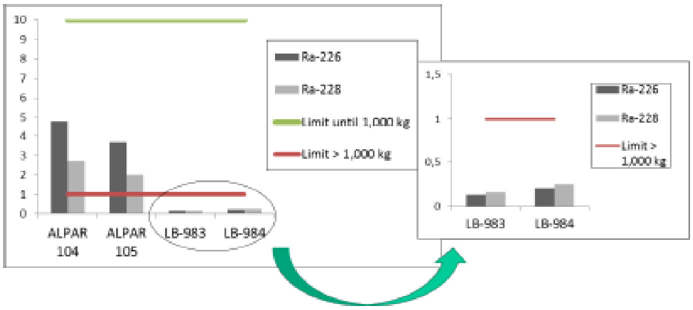

Fig. 1. Behavior of results in relation to Brazilian disposal limits.

Keywords: *NORM, Thermal Desorption, Dilution*

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ET6 (T1.A-0429)

Focus on Ethics in Radiological Protection for Medical Diagnosis and Treatment

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The International Commission on Radiological Protection (ICRP) set up a task group (TG 109) to advise medical professionals, patients, families, carers, the public and authorities about the ethical aspects of radiological protection of patients in the diagnostic and therapeutic use of radiation in medicine, in specific case of patient. Occupational exposures and research-related exposures are not in the TG 109 scope. The starting point of TG 109 is ICRP Publication 138, which identifies four core values (beneficence/non-maleficence, prudence, justice and dignity) and three procedural values (accountability, transparency, and inclusiveness) associated with the system of radiological protection. The goal of this TG is to explain them in the context of diagnostic radiology and radiation therapy, and put into perspective with the principles and values of biomedical ethics (typically: beneficence/non-maleficence, justice and autonomy). The patient's volition is an essential part of the acceptance of risk in medical exposure. This directly comes from the values of dignity and autonomy and is deeply associated with the principle of justification in radiological protection. The decision of accepting a risk is the issue of dignity and prudence, and usually based on the application of the beneficence/non maleficence values, which require careful assessment of risk and benefit. In many cases this is oversimplifying or impracticable, and other core and procedural ethical values need to be taken into account. Depending on the situation, the values of solidarity or common good, sustainability or the good of future generations, as well as honesty and empathy could also be useful in this process. Compliance with dignity and autonomy also means that when the evidence base is not conclusive, the uncertainties involved have to be disclosed, both to allow the patient to make a good decision and the give real informed consent. The 'right to accept the risk voluntarily' and 'an equal right to refuse to accept', was already mentioned in ICRP Publication 62. Together with the concept of right to know, informed consent was clearly established in ICRP Publication 84 on pregnancy and medical radiation. Once the theoretical framework has been presented, the approach taken by TG 109 takes a practical and pragmatic path. An evaluation method is proposed to analyse a specific situation, to develop criteria or for education purposes. A wide range of situations (e.g. pregnancy, elderly, paediatric, end of life) are considered in two steps: first within a realistic scenario on which the evaluation method is applied; and the second within a more general context. Scenarios are presented and discussed, with attention to specific patient circumstances, and on how and which reflections on ethical values can be of help in the decision making process.

Keywords: Ethical aspects of RP of patients, Reflection on ethical values in patient medical exposure

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ET6 (T3.D-0283)

Strengthening Radiation Safety Culture in Medicine, IAEA Training

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The number of early acute health effects and deaths attributed to radiation accidents in medicine exceeds the number from accidents in any other peaceful use of radiation including nuclear power.(1) There are a large number of incidents that have been reported and which resulted in detrimental patient effects from unintended exposure in interventional radiology,(2) nuclear medicine,(3) interventional cardiology(4) diagnostic imaging(5) and radiation therapy The need exists to prevent these detrimental effects that arise from medical errors or unintended exposure.

The approach to Radiation Safety Culture in Medicine brings together concepts from the "Just Culture" movement regarding personal accountability and respectful work environments; concepts from dose-reduction programs, such as Image Gently® and Image Wisely®, which emphasize conservative and thoughtful decision-making; and provides tools for applying the fundamental underpinnings of radiation protection – Justification, Optimization, and Limitation – in the context of radiation medicine.

The value of safety culture in medical applications has also been long-recognized. The "Bonn Call for Action(6) specifically identifies the strengthening of radiation protection safety culture as one of its core ten actions to improve radiation protection in medicine.

The training is provided as a workbook offered to radiation medical professional, members of the regulatory community and professional organizations. The information can be introduced into formal educational programs; as workshops; or as continuing professional development. The material is based on the 10 safety traits used in high complex technologies such as nuclear power and aviation. The traits and associated material provide the student with an understanding of the importance of the trait through case studies, a digital presentation of a positive example of the trait and a series of questions that support the understanding of the trait. This methodology was specifically developed for the adult learners through, visual, written and tactical activities. There is a corresponding trainers manual that helps reinforced the principles of adult learning.

The training material is being launched at the IRPA15 meeting and the material is freely available on the RPOP website and "train the trainer" programs are planned for 2020.

Keywords: safety culture, radiation protection Keyword1, Keyword2, Keyword3 (limit keywords to three words, italics, left alignment)

ET7 (T3.7-0362)

Stepwise Approach for Remediation of the Uranium Tailings in Central Asia

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There are ongoing and planned physical remediation activities of uranium legacy sites in Central Asian countries including Tajikistan, Kyrgyzstan, Kazakhstan and Uzbekistan. Average gamma dose levels at the sites range from 0.30 µSv/h to 4.0 µSv/h, which equals an exposure of between half an hours to four hours of average global natural background radiation. A number of factors could cause the contamination to spread and accumulate covering large territories. Since the planning of remediation activities requires consideration of all aspects of ensuring the efficiency and safety of the work being carried out, as well as affecting all phases of remediation activities, regulatory requirements should cover all elements of the planning of relevant activities taking into consideration local conditions.

Norwegian Radiation and Nuclear Safety Authority (DSA) has introduced the "Regulatory threat assessment" during the cooperation with Central Asian countries in 2009. It is a stepwise approach that includes the analysis of the existing regulatory situation in radiation and nuclear safety in a country and identification of gaps including those related to radioactive waste management and remediation. Elimination of the gaps through developing or improving new regulations taking into account the recommendations of all stakeholders involved in the process of remediation of uranium industry wastes. New regulations should use the results obtained during the studies of the possibility of processing and disposal of radioactive waste, the characteristics of the waste of the uranium industries as well as

To the date, there is only a general legislative framework in this area and no detailed regulations and legal acts related to the activities of the remediation of uranium legacy sites.
Therefore, it is difficult to ensure radiation safety of workers, public and the environment during remediation activities for both operator and regulatory body of the territories planned for remediation. Due to limited resources for remediation in Central Asia, DSA along with other international organizations putting its efforts related to better management and coordination through its regulatory assistance.

Keywords: remediation, DSA, regulations, stepwise approach.

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ET7 (T6.5-0474)

Planning and Preparing Waste Management after Severe Nuclear Accidents – a Methodological View

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Large releases of radioactive material into the environment inevitably produce large amounts of material to be safely removed, processed and stored. These materials include contaminated waste from decontamination work, agricultural products unsuitable for further use, contaminated goods and commodities, etc. Typically, these materials differ in type and amount from normal waste. They are not nuclear waste as they are not waste from a planned exposure situation and they are not "normal" waste due to their content of radioactive isotopes. Consistently, legal regulations which are tailored either for nuclear and non-nuclear industry are not suitable. Non-nuclear waste is not assumed to be contaminated at all and nuclear waste is typically of low volume.

Emergency preparedness must include planning for handling of such contaminated waste, as it would be too late to do it only after an accident. The outstanding challenge is the mere amount of material that potentially may be produced. It is hence a primary basic step in planning to assess the different types of material and their amount on the basis of adequate scenarios. A second step is a logical concept for setting a framework for handling this contaminated waste, including e.g. operational intervention levels and maximum permissible contamination levels for such material. One option would be scaling existing limits for exclusion and exemption for planned exposure situations using reference dose values adjusted to the emergency situation.

As an example, we assess the amount of waste from urban and agricultural decontamination, food production and contaminated conventional waste. The calculation assumes a major INES 7 accident and uses the RODOS decision support system. A large number of realistic weather conditions are considered in the assessment.

Amounts of bulk matter are assessed separately for different kind of waste, different levels of waste contamination and for areas with different levels of environmental contamination. This data allows for adjusting operational levels bearing in mind the consequences. On the other hand, potential pathways must also consider dose constraints to workers and population from processes and deposited waste.

A clear, consistent and transparent waste management concept is also needed in order to maintain trust and confidence after an accident. Investing in development of concepts and planning in the planning phase certainly helps to avoid mistakes and misperception in the emergency and recovery phase.

ET7 (T7.5-0355)

Progress in Addressing Challenges in Nuclear and Radiological Legacy Management

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Many countries with nuclear energy and related programs, involving man-made and naturally occurring radioactive material, are facing challenges with nuclear and radiological legacy sites and installations. They must be managed in an open and transparent way, addressing the views of relevant stakeholders, so as to build confidence in the solutions being developed, and through that developing process, that will help to avoid the creation of more legacies in the future. Common characteristics that bound the development of a practical approach to legacy site management and regulation include the following:

- Each legacy presents unusual features; typically, a complex combination of radiological, chemical, and physical hazards and other operational challenges.
- Radiological and other hazard characteristics are, initially, broadly unknown: appropriate and adequate records may have been lost or were never kept; former site operators with knowledge of the site are unavailable or site ownership has changed hands several times and responsibilities for the site are not clear, etc.
- Regulatory circumstances are complex because, for example, the site was not operated in line with current standards, recommendations and guidance, and the current regulatory framework was not designed to address these circumstances.

To contribute in solving these issues, an Expert Group on Legacy Management (EGLM) was given the mandate to assist Nuclear Energy Agency (NEA) member countries by preparing guidance on practical interpretation and application of radiological protection to legacy management, and support the development of corresponding regulatory guidance. The overall goal is to develop a practical and harmonized approach for the regulation of nuclear and radiological legacy sites and installations, taking into account the results of other relevant activities of the NEA, the International Atomic Energy Agency (IAEA) and the International Commission on Radiological Protection (ICRP), while accounting for good practice at different types of legacy sites, as illustrated by specific examples. To this end, the EGLM collated experience from 13 case studies and site visits from around the world covering a wide range of prevailing circumstances. The EGLM has then considered how to address the identified challenges under the following headings:

- Regulatory frameworks;
- Characterization of circumstances;
- Societal aspects;
- Deciding upon and achieving end-states, and
- Long-term protection values.

This paper will review the conclusions and recommendations of report of the EGLM (1) and discuss them in the light of continuing activities of the NEA's Committee on Decommissioning and Legacy Management aimed and related international developments.

Keywords: Decomissioning, legacies, NEA

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ET7 (T7.5-0360)

Application of the System of Radiological Protection to the Management of Contaminated Areas

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There is a number of situations which are characterized by the presence of radionuclides associated with past human activities that appear nowadays, for some reason, to be unacceptable for stakeholders and then of concern for the regulators who are responsible for the management of such contaminated areas. These areas, sometimes designated as legacy sites, cover a wide variety of cases, from large areas impacted by nuclear weapons testing to dwellings where watches were painted with radium paint. There may be research laboratories, nuclear research reactor, military complexes that were shut down but not remediated or at least not remediated according to current standards and societal expectations. Former uranium or rare earth mines may also fall in the scope of contaminated areas, as well as areas that were affected by accidental or uncontrolled radioactive discharges.

There is a large variety and thus circumstances to be considered while dealing with contaminated areas. In most cases, in addition to radionuclides, other pollutants such as heavy metals or chemicals need to be considered, requiring a holistic approach for risk management. Definition on an end state, stakeholders' engagement process, waste management, protection of the environment must be considered together with prevailing circumstances in order to reach and implement a sustainable remediation strategy.

This presentation describes the application of the System of Radiological Protection as described in ICRP Publication 103 to contaminated areas excluding post-accident situation following a severe nuclear accident. A number of case studies supports this application which reflects the work of ICRP Task Group 98.

Keywords: Contaminated areas, graded approach, stakeholders

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ET7 (T7.5-0370)

Regulatory Supervision and Radiation Survey in the Area of Former Military Technical Bases

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Introduction. In the early 1960-s, the Coastal Technical Bases of the Navy has been created in the Northwest Russia, behind the Arctic Circle. Coastal Technical Bases were used to support the reloading of nuclear reactors of nuclear submarines and temporary storage of the spent nuclear fuel (SNF). Also, liquid and solid radioactive waste (RW) being originated both from the activities of nuclear submarines, above-water ships with nuclear powered installations and nuclear service ships were stored at Coastal Technical Bases.

Materials and Methods. Since 2004, The State Research Center - Burnasyan Federal Medical Biophysical Center of Federal Medical Biological Agency in cooperation with the Norwegian Radiation Protection Authority carries out some research and practical works to regulate radiation safety at these Bases. The methods used: radiation health physics in field work, radiometry, dosimetry, including the use of personal dosimeters being exposed the personnel and in-situ, spectrometry, radiochemistry.

Results. The main directions of research are as follows: radiological threat assessment; detailed analysis of the radiation situation in the area, on-site, and in the vicinity of the Bases; radiation monitoring of the environment; development of computer maps and geo-information systems; expert review of projects and designs in the field of remediation, including SNF and RW management. The radiation situation at the industrial sites and within the health protection zone is characterized by significant local soil contamination, and this leads to the environmental contamination and potential spreading of radioactivity beyond the industrial site, including to the off-shore water area. The highest soil contamination due to manmade radionuclides is registered around the SNF storage facilities, where the specific activity of ^{137}Cs reaches 5.7×10^7 Bq/kg, and that of ^{90}Sr – 5.7×10^6 Bq/kg. The performance of the mentioned field, practical and theoretical works terminates with the development of a set of regulatory documents to assure observance of the radiation safety requirements for workers, the public and environment, as well as documents to regulate SNF and RW management on the sites.

Conclusions. The experience accumulated during remediation of the former Naval Coastal Technical Bases, has helped to identify new relevant areas of improvement of the regulatory supervision at nuclear legacy sites. When dealing with the protection of the population and environment, a methodology of comprehensive radiation and chemical monitoring should be developed. An important link of the social focus is to improve strategies of public communications near nuclear legacy sites under remediation

Keywords: radioactive waste; spent nuclear fuel; radioactive contamination; samples of environmental media; effective doses; regulatory documents

ET7 (T3.D-0283)

The Radiation Protection of the Population during Remediation of the Uranium Legacy Sites in the Russian Federation and in the Central Asia Countries

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Introduction. In the USSR, the main uranium mining and milling facilities were located in the Russian Federation and in the Central Asian Republics. After the breakup of the Soviet Union, the development of the majority of uranium ore deposits in these republics was partially or fully stopped. Currently, many facilities and sites in areas of former uranium development acquired the status of uranium legacy sites. Since 2005 and up to now we performed and continues researches at the sites of uranium facilities in Russia (The Priargun Mining Chemical Association in the Chita Region), and in the Central Asia countries (Tajikistan, Kyrgyzstan).

Materials and Methods. The main areas of research, being completed both in field and in laboratories, included: study of concentrations of radionuclides, using gamma spectrometry and radiochemistry methods in the environmental media and local foodstuffs; individual public and in-situ dosimetry; calculation and evaluation of external and internal doses to the population due to all radiation sources involved in this region.

Results. Studies being conducted at the uranium legacy sites of the Priargun Mining Chemical Association, revealed significant contamination on the territory of the Octyabrsky village, which is included in the health protection zone of the plant, as well as the very high values of individual effective doses to the population living in this village, mainly due exhalation of ^{222}Rn. Based on the findings of our studies the decision was made to resettle the residents (more than 4000 persons) to more radiation safe place. In the settlements located near uranium legacy facilities in Kyrgyzstan under remediation (Kaji-Sai and Min-Kush villages), the public exposure induced by the natural sources is increased, reaching 6.0 mSv/year. Close to the Taboshar tailings in Tajikistan (Istiklol City), dose induced by the natural components is much higher – 12.6 mSv/year. The key component with the very significant contribution to the effective annual public dose at the surveyed areas is exposure due to inhalation of ^{222}Rn decay products.

Conclusions. The researches at the uranium legacy sites have revealed the insufficiency of the available regulatory framework, therefore, we developed regulatory document – the Guidance "Radiation Safety Regulation during Remediation of the Former Uranium Mines". In addition, our experience in regulation of the public and environmental protection, being accumulated at uranium legacy sites, has found its reflection in a recent TECDOC, which includes the findings of activities of the IAEA International Forum on the regulatory supervision of legacy sites.

Keywords: uranium legacy sites, radioactive contamination; samples of environmental media; radon, effective doses; regulatory documents

ET8 (T5.2-0242)

Identification and Assessment of the Hazards in a Nuclear Fuel Fabrication Facility

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In uranium fuel fabrication facilities, large amounts of radioactive material are present in a dispersible form. This is particularly so in the early stages of the fuel fabrication process. In addition, the radioactive material encountered exists in diverse chemical and physical forms and is used in conjunction with flammable or chemically reactive substances as part of the process. Thus, in these facilities, there are radiological and chemical hazards will occurs during the operation. The main hazards in the nuclear fuel fabrication plant uses with uranium enriched LEU (20%) are potential criticality events. The other hazards in these facilities may be occurs the releases of uranium hexafluoride (UF6) and (U3O8), from which workers, public and the environment should be protected. In nuclear fuel fabrication facility, the process for the obtainment of U3O8 for fuel elements fabrication for research reactors starting from UF6 comprises two well defined stages characterized by the risks involved in the raw materials and intermediate products. The first stage is the wet process (conversion process) includes the hydrolysis of UF6 to UO2F2 and posterior precipitation to ammonium diuranate (ADU); the second stage is dry process to obtain the U3O8 powder from ADU at high temperature. This work presents the analysis of the events in nuclear fuel fabrication facility that have as a consequence the stated risks, their detection and prevention to protect the workers, public and environment from both radiological and chemical (toxicity material) hazards.

ET8 (T5.4-0613)

Mo-99 Contamination Incident Leading to Tissue Reactions to the Hands of a Radiopharmaceuticals Manufacturing Worker

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The Australian Nuclear Science and Technology Organisation (ANSTO) operates one of the world's most modern nuclear research reactors, OPAL; a comprehensive suite of neutron beam instruments; the Australian Synchrotron; the National Research Cyclotron; and the Centre for Accelerator Science. ANSTO also provides the Australian and international community with nuclear medicine including Tc-99m using Gentech® generators which contain the parent nuclide, Mo-99.

During August 2017 a radiopharmaceuticals manufacturing QC analyst became contaminated whilst carrying out standard operating processes with a QC sample of Mo-99. Despite the rapid reactions of the operator the radioactive contamination on the analyst's gloves and hands led to an estimated dose to the skin of the hands of approximately 20Gy. The accident was reported on the IAEA International Nuclear Event Scale (INES) as a level 3, serious accident. There were a number of improvements to equipment, process and protective equipment identified in the resulting investigation.

This presentation will describe the process which led to the accident, the immediate responses and causes of the accident, initial dose estimates and the error margins of the early dose estimates based on the information available at the time and lessons learned. Immediate changes to equipment were implemented and longer term modifications requiring significant design changes were identified to reduce the risk of a similar accident happening and these will be described. The tissue reactions due to the radiation exposure to the workers hands will be described along with the medical treatments administered. Finally, the emotional / psychological impacts will be briefly discussed.

***Keywords**: Contamination, Accident, INES*

ET8 (T5.6-0497)

Comparative Dose Rate Assessment for VVR-S Nuclear Research Reactor Hot Cells Decontamination

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VVR-S nuclear research reactor is under decommissioning due to safety reasons from 2010. Reactor was operated between 1957 and 1997 having as main purposes radioisotopes production for medical and industrial applications (in the Reactor Hot Cells) and research in physics, biophysics and biochemistry. Hot Cells decommissioning consisted of radioactive waste evacuation, radiological characterization of the internal parts (equipment's and stainless-steel lining), surfaces decontamination and dismantling and evacuation of the internal parts. The floors were contaminated as a result of the irradiated material spreading from damaged capsules or liquid vials. The radiological risk for the workers who performed floor surface contamination measurements (dosimetrist) and floor decontamination (mechanical worker) was assessed for the Hot Cell no. 4 due to the fact that the handling devices were damaged. Direct measurements of the ambient dose equivalent H*(10) were taken in order to identify the floor hot spots. Then, dosimeters with thermo-luminescent detectors (TLDS) tissue equivalent with high sensitivity were placed above the hot spots. From each hot spot, samples were taken and measured using a gamma-ray spectrometry system with a high-purity germanium coaxial detector (HPGe). Dose rate was estimated based on the hot spots activity concentration by using a standard calculation method as well as numerical method (RESRAD Build code). Dose rate calculation by standard method was performed assuming that the dosimetrist was exposed for 5 minutes during H*(10) measurement and located 70 cm away from the measurement point. The dose rate was 3.39 mSv/h, 0.28 mSv (5 min.) and the risk was quite high considering a dose limit of 20 mSv/year, 2000 working hours/year (10 µSv/hour). In such circumstances the working time must be less than 5.9 hours/year. The mechanical worker was positioned 45 cm away from the hottest spot and both external and internal exposed for 12 minutes. Dose rate was 7.97mSv/h, 1.59 mSv (for 12 minutes) and the risk is very high. The working time must be less than 2.5 hours/year. The internal committed effective dose was calculated assuming only 10^{-4} of the total floor activity was released in the Hot Cell atmosphere and that the mechanical worker wore a mask with a filter retention efficiency of 99%. The dose rate is relatively low (of 1.62 µSv/h) due to the high efficiency of the filter mask. The assessment of the dose rate for mechanical worker during floor decontamination with ResRad code, was performed considering a surface source (2 m radius circle) located in the middle of Hot Cell, 3 decontamination cycle (12 min duration/cycle), 3 distinct receptor locations. The assessment was performed initially, after 7 days, 14 days, 21 days and 28 days. The dose rate obtained using RESRAD Build code are 32 % lower than those evaluated with standard method due to the model complexity. Both methodologies used for dose assessment are in agreement and useful in similar exposure situations.

Keywords: Hot Cells, Decommissioning, Dose rate

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ET8 (T5.7-0106)

NORM Waste Management: Impact on Landfill Workers

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French regulators have taken advantage of the transposition of the European directive 2013/59/Euratom to update the regulation on the management of waste from industries involving NORM. The activity concentrations of waste containing NORM are highly variable depending on the raw materials used in the industrial processes and on the processes themselves, from a few Bq/kg to several thousand Bq/kg of uranium 238, thorium 232 and/or their daughters, as well as potassium 40.

One of the evolutions is that NORM waste whose activity concentrations is lower than 20,000 Bq/kg of uranium 238 and/or thorium 232 may be stored in a landfill for usual dangerous substances, and not necessarily in a radioactive waste storage facility.

The impact on workers of such a decision may be assessed on the basis of workplaces studies. To this end, it is necessary to identify the different workplaces concerning by exposure to NORM waste in the landfill. For each identified workplace, the assessor has to define exposure scenario(s) including degraded operating situations and the associated hypotheses as:

- the type of waste (pasty or powdery, rich in uranium 238 or rich in thorium 232 or in potassium 40),
- the exposure parameters to assess external and internal exposure (such as distance between the radiation source and the workers, the suspension rate, inhalation rate, etc.).

The aim of this presentation is to describe these scenarios and the associated hypotheses.
On the basis of the results, some recommendations may be made to the authorities (for inspection) and to the operator of the landfill (for workers protection).

Keywords: NORM, waste management, occupational exposure

RC

The System of Radiological Protection and Exposure Situations

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The International Commission on Radiological Protection (ICRP) develops the System of Radiological Protection, taking into account the latest scientific knowledge, ethical values, and practical experience. It is the basis of standards, legislation, guidance, programmes, and practice worldwide.

The objective of the System of Radiological Protection is to contribute to an appropriate level of protection for people and the environment against the harmful effects of ionising radiation exposure without unduly limiting the individual or societal benefits of activities involving radiation.

This short course is an introduction to the aims, scope, principles, key concepts, tools and requisites, and the scientific and ethical bases of the System of Radiological Protection.

It includes a brief review of radiation effects, quantities and units used in radiological protection, and the primary aim and specific goals of the System of Radiological Protection.

In particular, the three exposure situations introduced in The 2007 Recommendations of the International Commission on Radiological Protection (1) – existing exposure situations, emergency exposure situations, and planned exposure situations – are reviewed. Their relationship with the three categories of exposure – public, medical, and occupational – are described, as well how the various protection tools and requisites are used in each case.

At the end of the course, participants should have an understanding of how the System of Radiological Protection is structured, and how the main concepts relate to one another.

Keywords: radiological protection, general recommendations

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RC

Conceptual Basis of Graded Approach in Radiological Protection

Jaiki Lee

There are differences in benefit of value associated with the exposure according to the reason or situation of the exposure. For taking into account of these differences, applying graded approaches in radiological protection(RP) seems necessary and proper. Protection efforts proportionate risk is the principle of graded approach and optimization of protection.

The International Commission on Radiological Protection(ICRP) lay down the conceptual basis of graded approach in RP through its recommendations. The current system of RP is based on exposure situations; planned, emergency, existing situations. Also, three categories of exposure and exposed individuals are identified; occupational exposure of workers, public exposure of members of the public and medical exposure of patients. Depending on the exposure situations and categories of the exposed individuals, protection approaches and dose restriction criteria are adjusted accordingly.

Unfortunately, there are incompleteness of coverage, contradictions and confusions in interpretation of the current system of RP. Examples are: severe exposure of peoples due to a radiological event seems fit to neither planned, emergency nor existing situation; it is problematic to say a potential exposure is part of a planned situation; dose constraints apply to carers and volunteers of a biomedical research while regarding their exposures are medical one; requirements applied to the occupational exposure in a planned situation may apply to workers exposed to radon at work while stating that it belongs to the existing exposure situation; it is questionable to apply the same protection approach to all the remaining individuals other than occupationally exposed workers and medically exposed patients as a single category of 'members of the general public'.

Due to such complexities and ambiguities, we face difficulties in communicating the conceptual basis of RP with general public or media. Sometimes, understanding of the current system differs even among experts. In this class, we will discuss the possibility of making the system better.

RC

Residential Radon Lung Cancer Risk

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A critical overview is given of the principal approaches (epidemiological and dosimetric) that have been used to estimate the lung cancer risk to the general population due to residential exposure to radon and its progeny in their homes. Starting with the findings of the epidemiological studies of underground miner cohorts exposed to radon their extrapolation to public exposure is described. An account is then given of the results of residential case-control studies of the general population and of the European, Chinese and North American pooling of such studies. Sources of radon exposure misclassification in general population case-control studies and their potential impact on radon risk estimation are described.

A summary is then given of model based methodologies used to make a global estimate of lung cancer mortality due to residential radon exposure. In the case of the dosimetric approach the implications for radon lung cancer risk estimation resulting from recent ICRP recommendations that doses from radon and its progeny should henceforth be estimated on the basis of dosimetric and biokinetic models are discussed. Finally a brief account will be given of some recent attempts to identify biomarkers specific to radon and radon progeny exposure in humans.

RC

Late Health Effects of Atomic Bomb Radiation: the Life Span Study of Atomic Bomb Survivors and Studies of Survivors' Offspring

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Epidemiological studies investigating late health effects of atomic bomb radiation and its transgenerational effects have been conducted among atomic bomb survivors (the Life Span Study: LSS), *in utero* survivors, and the children of the survivors who were conceived after parental exposure to radiation (F_1). Health examinations for subcohorts of the LSS and *in utero* survivors are conducted as the Adult Health Study (AHS) and a program of health examinations for a subcohort of the F_1 study is called the F_1 Offspring Clinical Study (FOCS). Results of these studies have improved our understanding of radiation pathogenesis and informed practices of radiological protection as well as improved health administration among atomic bomb survivors.

These studies have observed increased radiation risks for malignant diseases among survivors including those exposed *in utero*, and possible risks for some noncancer diseases. A marked excess of leukemia cases emerged in the early period, while the risk of solid cancers increased later and continues to be elevated today. The risk of all solid cancers was approximately 40 to 50% per Gy higher than that of the non-exposed baseline for both mortality and incidence for a person aged 70 years who was exposed at age 30. The dose-response is linear, but a significant concave curvature has appeared recently for men. Reasons for this are under investigation. Effect modification by age at exposure, implying age-dependent radiosensitivity of the tissue, generally indicated that younger ages at exposure induced higher radiation risks. Breast and uterine corpus cancers showed the highest radiation risks at ages at exposure around puberty, during which those organ tissues rapidly grow. These findings suggest that tissue stem cell activity may be related to susceptibility to radiation carcinogenesis.

Risks of heart disease and stroke increased with radiation dose as a whole but the association varied by subtype in the LSS and AHS. Immunosenescence and persistent inflammatory reactions, which were observed among the survivors who were exposed to relatively high-dose radiation, might accelerate vascular pathogenesis. Cataracts were associated with radiation exposure. As for thyroid diseases, nodular diseases were associated with radiation dose, but thyroid dysfunction or autoimmune disease was not associated. Cognitive dysfunction or dementia did not seem to be accelerated by radiation exposure. As pathogenesis of radiation-associated noncancer diseases is still under investigation, especially for those at low doses, study results need to be carefully interpreted.

In contrast, no increased risks due to parental exposure to radiation have been observed for malignancies or other diseases in F_1 at present, but continuing investigations are required.

Keywords: Atomic bomb, Cancer, Noncancer disease

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OS1.1 (T1.1-0157)

The Radiobiological Human Tissue Repository at the Southern Urals Biophysics Institute

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The Radiobiological Human Tissue Repository (RHTR) at the Southern Urals Biophysics Institute was established more than 20 years ago and has been supported with the research collaboration of the US Department of Energy and the Russian Federal Medical Biological Agency [1]. The RHTR collects, processes and stores biospecimens from workers of the first Russian nuclear production facility, Mayak PA, who were occupationally exposed to ionizing radiation at a wide dose range, from offspring of the workers and also from individuals (not occupationally exposed) who live in the region close to the facility.
To date, the RHTR collection includes 324,742 biological specimens contributed by 10,311 individuals (aka, registrants); each specimen is assigned with a unique identification number that is printed on a barcode label of each stored specimen. The RHTR collection is divided into five sub-banks of biospecimens: the Autopsy Tissue and Organ Bank contains 154,505 specimens from 1062 registrants; the Surgical/Biopsy Tissue Bank contains 2702 biospecimens from 991 donors; the Repository of Blood and Blood Components contains 130,435 biospecimens contributed by 7940 individuals; the Genetic Material Bank contains appr. 45,000 biospecimens from 4750 donors; and the Repository of Other Tissues (e.g. sputum, or immortalized B cells from selected individuals).
All biospecimens are annotated with complete demographic, medical, dosimetry and occupational information about donors. Data on non-radiation factors that could affect a registrant are also available.
The electronic database of the RHTR contains information on the stored biospecimens and their locations, and generates an individual barcode for each item in storage. Scanning a barcode label provides complete information on a donor of a specimen and enables fast and precise control of its location and any transfers within the repository or to outside researchers.
The RHTR web site created under the collaborative efforts (rhtr.subi.su) is based on the electronic database that delivers to the web-site de-identified metadata that can be requested by prospective users. Interested researcher can look through the catalog of biospecimens and available annotated information in real time using the web site. To date, the RHTR is the only biorepository of the kind in Russia and one of the few biorepositories in the world that features real-time metadata searching.
The stored biological specimens and complete medical and dosimetry annotated data on donors of the collected samples are a unique resource for investigations of mechanisms of radiation–induced health effects following chronic radiation exposure at low dose rates.

Key words: repository, biological specimens/samples, nuclear worker cohort, Mayak PA

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OS1.1 (T1.C-0234)

Forging a Sustainable Global Alliance for Radiation Research and Education

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The field of radiation research is highly specialized, extremely interdisciplinary, with a body of knowledge that is rapidly shifting. It also represents an increasingly fragmented and shrinking population that is progressively isolated. Participation in related technical societies such as the Radiation Research Society and the Health Physics Society have been declining, accompanied by a 'greying' of remaining members. Fiscal pressures push universities to cut small programs, such as radiation science, in favor of larger ones. Costs motivate employers to 'train' replacements when knowledgeable workers retire. Because the field of radiological sciences is so small, it will not survive as an academic discipline, unless concrete steps are taken. Initiatives meant to address the decline of radiation professionals have failed, because sustainability of the entire ecosystem (education, training, research and the workforce) has not been addressed. It is essential to create a private/public global partnership to sustain education and research in the radiological sciences. This presentation is intended initiate a discussion amongst academic institutions, radiation protection scientific advisory bodies, regulators, industry, and medicine. The objective of this discussion is to lay out a framework for forging a global alliance that sustainably supports research and education in the radiation sciences. A new type of knowledge ecology is needed to connect academic institutions, professionals, associations, and industry with a focus on sustainability of the profession. A network is needed to provide a continuous means of knowledge sharing, experimentation, application, feedback and improvement that will be able to cross both geographic and cultural borders. A sustainable educational alliance can be built through increased use of virtual educational content, internationally collaborative field research, and coordinated student learning opportunities. Admittedly, such alliances will need to address issues such as sharing curriculum amongst academic institutions; developing joint, dual or certificate degrees; providing field and industry research experiences for students; sustaining funds for research grants and student support; maintaining intellectual property rights; and facilitating knowledge transfer between institutions. Equally as important, sustainably requires understanding and matching the number of skilled graduates with the needs of employers. Wages must be sufficient to attract students into the discipline and retain them in the workforce. Too many graduates forces downward pressure on wages. Too few students results in program closure. Viable academic programs require a combination of student population, scholarships, fellowships, and funded research. This presentation is intended to initiate a discussion to build the Research and Education Alliance for Global Radiation Protection (REAGRP).

OS1.1 (T1.C-0521)

Global Networking for Low Dose Research

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One of the key radiological protection research areas addresses the need to improve our understanding of radiological health risks that might be caused by exposure to low radiation doses (meaning below to far below about 100 mSv). This vast subject includes addressing such aspects as chronic *versus* acute exposures, effects of dose level and dose rate, effects of different types of radiation, organ and tissue sensitivity, cellular damage mechanisms, tumour progression pathways, non-cancer adverse outcome, etc. Given the importance placed on such research by government funding organisations, and recognising the enormous amount of research done and continuing in this area across the globe, while noting ongoing efforts to effectively collaborate and coordinate research, it is felt that the global nature of ongoing work merits consideration of some level of global coordination. To address this need, the OECD Nuclear Energy Agency (NEA) has organised a "High-level Group on Low Dose Research" (HLG-LDR), with the aim of building a network of research Funding Organisations, Research Organisations, Regulatory Organisations, International Organisations, and Stakeholders, to facilitate the global coordination of low-dose research in response to societal challenges. The vision is that the HGL-LDR will support radiological protection policy, regulation and application choices by improving the effectiveness and efficiency of research through global networking for the coordination of ongoing and future low-dose research projects. This presentation will describe the approach and the foreseen programme, as well as how this global networking initiative will build upon existing research association worldwide (*e.g.*, European Research Platforms such as MELODI, the International Dose Effect Alliance, PLANET and J-MELODI in Japan, and others).

Keywords: Low dose research, global network, radiological protection policy

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Adverse Outcome Pathway Approach to Low Dose Radiation Effects

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A large body of biological research has been generated over the past century using in vitro, animal and epidemiological models. This data represents efforts globally to understand the mechanistic basis of radiation-induced health effects. However, it has remained difficult to effectively incorporate this data to derive meaningful information for refining guidance on the radiation exposure to the public.

In the last several years, an international agreement has been growing to support a paradigm shift from a "stressor" centric to an "adverse outcome" approach to risk management. In chemical and ecological toxicity such a tactic has been developed that integrates complex biology to apical endpoints through a set of key events, and key event relationships. The approach offers a means to capture available mechanistic knowledge in the literature and link it to outcomes of relevance to chemical toxicity, namely the adverse outcome pathway (AOP). The Electric Power Research Institute, Health Canada, U.S. Department of Energy, U.S. Environmental Protection Agency, Organization for Economic Cooperation and Development (OECD), and the OECD Nuclear Energy Agency (OECD/NEA) have been cooperating to bring this approach to the field of low dose radiation research. The current status of the AOP framework in the chemical field, how it can be applied to the radiation field, the EPRI International Dose Effect Alliance, the OECD/NEA High Level Group for Low Dose Research, and a vision for the next steps will be discussed. For this approach to be successful, the radiation research community will need to come together to assess the literature and help harness this data in a systematic manner for incorporation into the AOP framework. It is anticipated that AOPs will be adopted as a method to inform issues of high to low dose extrapolation, have utility in supporting low dose radiation risk assessment, and create a paradigm within which radiation research priorities may be developed.

Keywords: Adverse outcome pathway framework, radiation risk assessment, linear no threshold, key events, adverse outcome

OS1.1 (T3.7-0695)

First Joint Roadmap for Radiation Protection Research

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Since the High Level and Expert Group (HLEG) report ten years ago, a remarkable reorganization of the European radiation protection research landscape has taken place. The HLEG report on European Low Dose Risk Research led to the establishment of the MELODI platform, an association of European institutes committed to low dose risk research. Almost in parallel the platforms ALLIANCE, addressing research on radioecology, and NERIS, on nuclear and radiological emergency preparedness and response were established, succeeded by EURAMED on medical radiation in 2016. The European Dosimetry Group (EURADOS) was already founded in the 1980's. The newly established SHARE platform brought in yet another research community with expertise in social sciences and humanities that aims to improve links between research and society. All these platforms are openly sharing their vision and Strategic Research Agenda (SRA) with the multidisciplinary scientific community.

While the individual platforms have brought together European scientists and consolidated their research strategies, there are also joint research interests shared by the platforms, in response to global societal needs and for these a joint roadmap is being developed. The scope of research planned in the joint roadmap is in the context of various existing and potential exposure scenarios, relevant from a societal and radiation protection point of view. The key aim is to provide answers to open questions related to the exposure of humans and the environment, the reduction of uncertainties in risk assessment and the provision of sound, applicable solutions for risk reduction where needed. Proposed solutions will have to be practicable and widely accepted by the scientific community and the public.

The progresses in research challenges described in the joint roadmap will have an impact on the radiation protection of humans and the environment in many ways. First, it will support the implementation of the European Basic Safety Standards, to help to cope with the new requirements and harmonize practices throughout Europe. Our holistic approach covers both risk assessment and risk management, as well as the development of tools, methods and best practices to cope with the issues related to radiation exposure, thus making a major impact on society. Research is needed for risk prediction and foresight. New knowledge will contribute to evidence-based recommendations at international level and informed risk communication. Research on risk management will help in risk reduction/prevention, improve the resilience of societies for emergencies, help to set up action plans and work on mitigation and remediation. Authorities operating in radiation protection, public health and environmental protection will benefit from investigations guided by the roadmap. Guides, recommendations and regulations are needed, along with good practices and reliable methods for field and laboratory work, and epidemiological investigations. A graded approach to risk management is needed, and research will help to place exposures and risks in perspective. Technological development generates new standards, technological innovations and improved capabilities.

The implementation of the Joint Roadmap is expected to have direct and widespread impact on radiation protection in Europe. In particular, the research foreseen, and the derived recommendations, will promote consolidated, harmonized and robust decision making in the field of radiation protection throughout Europe and beyond.

Keywords: *Joint roadmap, radiation protection research, research challenges, societal impact*
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Childhood Leukemia around the Belgian Nuclear Sites, 2006-2016: An Ecological Study at Small Geographical Level

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The possible health risks associated with living around nuclear installations have been the object of public concern for several decades. Several epidemiological studies, particularly in European countries, have investigated the incidence of childhood leukemia around nuclear sites. In Belgium, the Belgian Minister of Social Affairs and Public Health commissioned, in 2008, a study to assess the possible cancer risks for populations living near the four Belgian nuclear sites of Class I (two nuclear power plants and two sites with a combination of industrial and research activities). Higher incidence of childhood leukemia was observed around one nuclear site with research activities for the period 2002-2008 (1). These analyses were however hampered by the geographical scale (e.g. the communes) at which cancer incidence data was available till then in Belgium and bias due to exposure misclassification could not be excluded. The present study investigates whether there is a higher incidence of childhood leukemia around the Belgian nuclear sites, by means of an ecological study performed at the level of the statistical sectors, the smallest Belgian geographical unit.

The analyses included 808 incident cases of childhood leukemia (aged < 15 years old) registered from 2006 to 2016 by the Belgian Cancer Registry. The distance between the centroid of the statistical sector and each of the four Belgian nuclear sites was calculated to define the proximity area. Rate ratios of childhood leukemia incidence were estimated using a Zero-Inflated Poisson model. Further analyses were performed to test the hypothesis of a gradient in childhood leukemia incidence with increasing levels of three surrogate exposures (e.g. distance to the sites, wind direction frequency and modelled hypothetical radiological discharges from the sites).

The Zero-Inflated Poisson model was adjusted for age, sex and socioeconomic status. A statistically significant higher rate ratio of childhood leukemia incidence was observed in the 5 km proximity area around one nuclear site with combined industrial and research activities whereas no significant higher rates were observed around the other sites. In addition, significant gradients for childhood leukemia incidence were observed with the different types of surrogate exposures considered in the 20 km area around this site. Further adjustment for potential confounding environmental exposures did not change the results.

This study confirms earlier results and suggests a higher incidence of childhood leukemia around one of the Belgian nuclear sites. Acute leukemia is the most common cancer type diagnosed in children but the incidence rate is low and it has to be stressed that these results are based on a very low number of cases. Further investigations into these findings could be useful to provide information at the individual level and to allow inferring causal relationships on the origin of the observations. More in general, this work stresses the importance of a better understanding of the etiology of childhood leukemia.

Keywords: Childhood leukemia, Incidence, Nuclear sites

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OS1.2 (T1.1-0564)

An International Cohort of Uranium Miners: The Pooled Uranium Miners Analysis (PUMA)

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Introduction: Previous individual and pooled epidemiological studies of underground miners have provided clear evidence that increased risk of lung cancer death is associated with higher cumulative radon exposure. These studies contributed to radiation protection measures limiting exposure to radon and its short-lived progeny at higher levels. However, questions remain regarding the risk of lung cancer associated with chronic exposure to lower levels of radon present in occupational and environmental settings. Furthermore, knowledge gaps for the risks of cancers other than lung and of non-cancer diseases, such as circulatory and respiratory system diseases remain. Therefore, the setting up of a large combined cohort gathering some of the most informative cohorts of uranium miners will allow us to conduct additional epidemiological analyses with increased statistical power and harmonized methods.

Methods: The Pooled Uranium Miners Analysis (PUMA) combines seven cohorts of uranium miners: three European cohorts (Czech Republic, France and Germany), two Canadian cohorts (Eldorado and Ontario cohorts) and two U.S. cohorts (Colorado Plateau and New Mexico cohorts). All included miners have available data on demographic and employment history information, vital status (date of death, and cause of death), and annual estimates of radon exposure. Some cohorts also have individual information on smoking history and/or external gamma radiation exposure.

Results: The PUMA study includes 124,507 uranium miners, corresponding to 4.51 million person-years of observation. In total, 54,462 deaths were observed, including 17,085 deaths due to all sites of cancer, 7,825 deaths due to lung cancer, 18,416 deaths due to circulatory diseases and 4,621 deaths due to respiratory system diseases. The mean durations of the follow-up of individual cohorts range from 30 to 39 years. The mean cumulative exposure to radon varied from 31 to 580 Working Level Months (WLM). Planned research topics include analyses of associations between radon exposure and mortality due to lung cancer, cancers other than lung, non-malignant diseases, and of modifiers of these associations, and characterization of overall relative mortality excesses and lifetime risks.

Conclusion: PUMA represents the largest study of uranium miners conducted to-date and provides opportunities to evaluate new research questions and to conduct analyses to assess potential long-term health risks associated with uranium mining that have greater statistical power to detect health risks at lower levels of exposure than can be achieved with any single cohort.

Keywords: Uranium miners, Radon, Mortality risk

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Genetic Markers Associated with the Development of Stochastic Effects

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One of the most debated issues is the low dose effect of ionizing radiation. Lately there appeared new data that testify to the increase in the risk of cancer and a number of non-cancer disease development following the exposure at doses up to 100 mSv [1]. Main system of cell protection includes DNA reparation, cell cycle control and apoptosis, antioxidant and immune systems. It is known that human population is genetically heterogeneous that is why the efficiency of work of the systems that maintain the genome integrity may differ greatly from person to person. It means that the response to radiation exposure and individual radiosensitivity may depend not only on sex, age and health status, but also on genetic peculiarities of a particular human being.

This study involved investigation of the connection between SNP genes that regulate reparation, apoptosis, cell cycle, antioxidant and immune systems (28 polymorphic regions) and risk of malignant neoplasms development. Expression of mRNA of *ATM, TP53, MDM2, CDKN1A, BAX, BCL-2, XPC, OGG1, STAT3, GATA3, MAPK8, NF-kB1* and *PADI4* genes was also studied in chronically exposed individuals.

Evaluation of gene expression and SNP studies were performed in 632 chronically exposed members of the Techa River Cohort for whom the dose to RBM had been reconstructed using TRDS2016 [2]. The association of SNP with the risk of malignant neoplasm development was investigated in 248 patients (median exposure dose to RBM was 0.87 Gy, dose range 0.002-4.6 Gy) with various types of solid cancers in the past history: breast cancer (49 people); skin cancer (46 people); GIT cancer (45 people); uterus cancer and cancer of ovaries (32 people); lung cancer (26 people). Comparison group consisted of 384 people without cancer (median exposure dose to RBM was 0.86 Gy, dose range 0.001-4.2 Gy). Gene transcription activity was studied in 163 people (median exposure dose to RBM was 0.72, dose range 0.08-3.51 Gy). Comparison group consisted of 146 people residing in the same settlements with cumulative dose to RBM less than 7 cGy.

The preformed study revealed the connection between polymorphic region rs1952133 of the *OGG* gene, rs2279744 of the *MDM* gene, rs2279115 of the *BCL2* gene, rs361525 of the *TNFα* gene and rs1050450 of the *GPX1* gene and increased risk of cancer development; and association between polymorphic region rs13312840 of the *NBS1* gene and rs1801270 of the *CDKN1A* gene with decreased risk of cancer development.

The most pronounced changes of the transcriptome in exposed persons were noted in genes that regulate apoptosis of *BCL-2, BAX, NF-kB1* and *PADI4*. Gene activity was decreased in exposed individuals with exposure doses to RBM >1 Gy.

The genes that we have studied play the key role in the system of cell protection from radiation exposure. Radiation-induced changes in the transcription activity as well as the presence of tumor-associated allelic variants of genes may impede effective work of protective mechanisms of a cell, which in its turn results in the development of late effects of radiation exposure.

Keywords: SNP, biomarker, chronic radiation exposure.

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Lifetime Risks in Cohort Studies of Uranium Miners

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The presentation includes models of excess relative risk derived from Czech, German and Ontario uranium miners developed for the most recent UNSCEAR report on Lung cancer from radon [1]. The three studies of uranium miners are the largest cohorts of uranium miners with 6349 lung cancers, more than twice than in the BEIR VI report (Table 1) [2]. Generally, the risk models based on low exposure or low exposure rates are preferred because of more reliable exposure estimates in later periods. The standard projection models suggested in BEIR VI report include modifying effects of attained age, time since exposure, and exposure rate. In addition to model BEIR VI, the presentation will include an alternative model where the modifying effect of exposure rate is considered in dependence of actual annual low (<0.5WL) and high (>0.5WL) exposure rates in contrast to mean exposure rate used in model BEIR VI. The second model uses all miners exposed to low exposure rates, whereas in the BEIR VI model the lowest category uses miners exposed only to low exposure rates and miners exposed to both high and low exposure rates are ignored. Higher numbers of cases in the latter model thus provide narrower confidence intervals in comparison to categories defined by mean exposure rates. The risk models will also be compared to 11 studies used in BEIR VI. The lifetime risks (firstly suggested in ICRP- 65) are calculated in relation to projection excess relative risk models using ICRP-103 background mortality tables for combined male and female European-American and Asian populations [3, 4].

Table 1. Cohorts of uranium miners involved in lifetime risk calculations

Cohort	Number of miners (thousands)	Number of lung cancers	Lifetime excess risk per WLM
Wismut	59	3942	0.00040
Ontario	28.5	1246	0.00083
Czech	10	1161	0.00056
11 cohorts of BEIR VI	60.6	2674	0.00087
7 cohorts of ICRP-65	31.5	1047	0.00028

Keywords: *Lung cancer, Radon exposure in uranium mines, Lifetime risk*

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OS1.3 (T1.2-0144)

Study of the Changes in the Lens during Cataract Development in People Exposed due to Radiation Incidents in the Southern Urals

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Until present day there is no clear understanding of the health risk values and nature of dose dependence for the population chronically exposed in the low to medium dose range (up to 1 Gy). In view of this, the studies in large cohorts could be very useful in providing necessary information.

Objective: To study the peculiarities of the lens opacities in exposed persons long after the chronic radiation exposure with account of the exposure dose effect and non-radiation factors. This case-control study involved patients included into the URCRM Registry of Exposed Population (1,377 persons) who were examined by the ophthalmologist in 2016-2018.). Analysis of the changes in the lens was based on LOCS - Lens Opacity Classification System, 3rd revision. It allows registering changes in the lens in various layers, including changes in the posterior capsule. One and the same method was used to examine and photograph lens opacities in all the subjects.

In the framework of this study for the first time individualized exposure doses to the lens were calculated using dosimetry system TRDS 2016.

The performed study in the cohort of exposed people, who for many years had been affected by low dose ionizing radiation, revealed that cataract incidence rate increases with attained age. The greatest number of cataracts were observed in persons aged >60, which is in line with trends for unexposed population.

The study of the lens changes in various layers revealed dose-dependence of the increase in the risk of opacity development in the posterior capsule (OR=1.54 (95% CI 1.04-2.27) and in the lens nucleus (the findings of 2 case-control studies demonstrated similar OR estimates (1.62 (95% CI: 1.01-2.60) and 1.84 (95% CI: 1.14-2.95)).

No statistically significant dependence on increasing dose of either lens changes in the anterior capsule and cortical layers or changes in the lens nucleus color was observed. Any impact of ethnicity on priority development of opacities in any lens layers was not proved either.

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A Multi-omics Approach to Assess the Effects of Low-Dose Chronic Exposure to uranium

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Human populations can be exposed to low doses of uranium either naturally present in earth crust or dispersed by human activities, such as phosphate fertilizers spreading, nuclear fuel exploitation and military weapons use. In this environmental context, social and scientific questions arise regarding such exposure and associated health risks.

To study the biological effects of such chronic low-dose exposures, an *in vivo* protocol of chronic exposure to uranium was developed and implemented. Male and female rats were exposed to a non-nephrotoxic solution of uranyl nitrate for nine months through their drinking water followed by analyses of metabolomic, transcriptomic (mRNA and microRNA) and global DNA methylation changes within a multiscale high-throughput multi-omics approach as a supplement to a clinical monitoring.

Results demonstrated a sex-interaction effect of uranium in the kidney, urine, and plasma metabolomes of rats. In urine and kidney, metabolic profiles confirmed that the primary impacted metabolisms were those of nicotinate-nicotinamide and of unsaturated fatty acid biosynthesis. Upstream of the metabolism level, transcriptomic analysis of the kidney revealed changes in gene activity associated with gene regulation mechanisms, cell signaling, cell structure, developmental processes and cell proliferation. Post-transcriptional epigenetic regulation revealed 70 dysregulated micro-RNAs; however no impact on DNA methylation level was observed. Further multi-omics analyses revealed the biological processes that were affected by exposure to uranium on multiple scales from gene expression to metabolome.

In conclusion, our results demonstrate molecular links between omics profiles (from gene expression to metabolism) of rats exposed to chronic low doses of uranium. Therefore, this multi-scale approach can be useful to elucidate effects of low-level environmental exposures systemically and mechanistically at multiple levels of biological organization. Finally, this approach may be useful for addressing a variety of radiation protection research issues, such as revealing mechanisms of delayed and even transgenerational effects, assessing individual radiosensitivity in the context of age, sex and state of health, as well as providing information for building Adverse Outcome Pathways for radiological risk assessment.

Keywords: uranium, low-dose, multi-omics

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OS1.3 (T1.2-0267)

"Southern Urals Population Exposed to Radiation" Cohort: Risk of Mortality from Circulatory System Diseases

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This report is devoted to the analysis of the effects of the chronic population exposure at low-to moderate doses and dose rates on the example of the risk analysis of mortality from diseases of circulatory system (DCS) in a newly established cohort of the population exposed in the Southern Urals.

In the 1950s in the Southern Urals residents of the Techa riverside villages were affected by chronic radiation exposure following the releases of Mayak PA radioactive waste into the Techa River. And the thermal explosion of the radioactive waste storage reservoir in September 1957 resulted in formation of the East Urals Radioactive Trace (EURT) and exposure of the population of the settlements located along the EURT.

Long-term follow up of the population exposed due to these 2 radiation incidents in the Southern Urals, uniform methods of follow up and analysis, common dosimetry system for dose estimation allowed combining this population into a uniform cohort for the analysis of radiation effects of low-to moderate exposure doses. Cohort of Southern Urals Population Exposed to Radiation" (SUPER) includes more than 60,000 people, the number of person-year at risk over the period 1950 -2015 was 1,836,203. The number of deaths from all DCS registered in the cohort is 14,830, and 6,163 deaths - from ischemic heart disease (IHD). Mean dose to muscle tissue calculated with the dosimetry system 2016, accumulated in 1950-2015 was 34 mGy, maximum dose was 995 mGy.

Excess relative risk was analyzed with simple parametric model (software package EPICURE). Available non-radiation factors for mortality risk analyses were time-independent (sex, ethnicity, catchment area, evacuation, radiation accident, parental exposure) as well as time-dependent ones (attained age, age at exposure, birth cohorts, calendar period, time after the exposure). We could not estimate the influence of specific factors that contribute to the development of cardio-vascular pathology, still, we believe that these factors will not change radiation risk value as they do not depend on dose of accidental exposure.

Risk analysis of mortality from DCS in SUPER cohort over a 66-year follow up period demonstrated dose-dependence of the mortality rate for all DCS and for IHD separately. Dose dependence is linear. ERR value (0.36/Gy for all DCS and 0.56/Gy for IHD) is comparable with the findings of other cohort studies [1-3].

Thus, the established cohort is highly potential for the future research of the risk of cancer and non-cancer effects related to chronic radiation exposure of the population.

Key words: ERR, diseases of the circulatory system, chronic exposure

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OS1.3 (T1.2-0392)

Cardiovascular Risks from Low Dose Radiation Exposure Review and Scientific Appraisal of the Literature

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In the past, the health risks of low-level exposure to ionizing radiation were assumed to be related primarily to cancer. However, it has been documented that at high radiation doses, a variety of other non-cancer effects have been observed, in particular, damage to the structures the cardiovascular system (particularly the heart and to the coronary, carotid, and other large arteries). An association between lower doses (< 0.5 Gy) and late circulatory disease has recently been suggested and remains somewhat controversial. The Electric Power Research Institute (EPRI), Radiation Safety Program, Low Dose Health Effects Research Focus Area has undertaken an evaluation of the scientific literature that includes recent epidemiological studies and mechanistic evaluations of the radiation effects on the circulatory and cardiovascular systems in order to assess the scientific bases of recommendations on lowering the cardiovascular disease nominal thresholds.

The EPRI research approach utilizes a systematic literature assessment to examine methodological strengths and weaknesses of epidemiological studies, resulting in a tiered stratification of the available studies, and indications of those studies most informative to a meta-analysis of results. For cardiovascular disease, a total of 71 studies were identified, with 20 of those studies being judged as Tier 1 (most informative) and 27 being Tier 2. All Tier 1 and 2 studies provided risk ratios for a given dose. The studies were then utilized in a meta-analysis assessing all cardiovascular disease, ischemic heart disease, and cerebrovascular disease. The resulting analyses indicates that a large uncertainty in risk estimates persists as evidenced by mixed results and wide confidence intervals. Nevertheless, increased risk may be associated with ionizing radiation at doses lower than previously assumed. The overall excess relative risk per gray was positive and significant for all cardiovascular disease and ischemic heart disease. The excess relative risk per gray was positive and significant for cerebrovascular disease only when acute dose studies were included in the analysis. The atomic bomb study dominates the meta-analysis when acute doses are included. Typically, the Mayak study is the dominating study when acute dose studies are not included, and the INWORKS study dominates when Mayak results are not included. An analysis for threshold was not possible with the available data.

***Keywords:** Cardiovascular Disease, Low Dose Ionizing Radiation, Meta-analysis*

OS1.4 (T1.1-0224)

Iodine Thyroid Blocking (ITB) in Case of Repeated Exposure to Radioactive Iodine: Preclinical Studies and Evolution of the Marketing Authorization (MA)

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Safe operation of a nuclear power plant (NPP) integrate emergency preparedness and response plan in case of emergency situations. These plans should be exercised and are implemented with the lesson to be learned from former incidental or accidental situations. Accidents occurring in the reactor core of a NPP may result in the generation of large amounts of radioactive particles, including iodine-131, dispersed as aerosols into the environment. These particles and aerosols potentially contaminate the affected areas, by deposits and can be inhaled by populations exposed to the radioactive plume. One of the main risks of the exposure to radioactive iodine is the occurrence of thyroid cancers, especially in infants and young children. In order to prevent exposure to radioactive iodine, three types of population protection countermeasures can be implemented: evacuation, sheltering and ITB. The aim of this last measure is to saturate the thyroid gland with non-radioactive iodine in order to prevent the fixation of the radioactive iodine of the plume. In France, the "iodine doctrine" advocates the single taking – renewable once in the adult individual in case of impossibility of evacuation of populations – of potassium iodide (KI) tablets which, to be effective, must take place between two hours before exposure and six hours after. However, the current Marketing Authorization (MA) for KI tablets does not consider the situation of populations or first responders potentially exposed to repeated and / or prolonged releases of radioactive iodine. Two questions arise regarding i) the conditions of repeated intake of stable iodine (side effects, frequency of setting, etc.), and ii) from a regulatory point of view, the possibility of changing the MA of the KI tablets, which for the moment only provides for a single dose, which may be renewed once.

The PRIODAC project (repeated prophylaxis by stable iodine in accidental situation) aims at refining ITB posology in case of repeated or prolonged accidental releases of radioactive iodine. The objective is to determine the optimal dosage and frequency of stable iodine administration to protect the thyroid gland, as well as to evaluate the safety of repetitive administration for different population's categories (infants, children, adults, pregnant women and the elderly).

Since the implementation of the program, the optimal effective dose (1mg/Kg/day) for a daily administration over a period of 8 days was identified and offer efficient long term protection of the tyroid. Preclinical experiments (clinic biochemical, endocrinology, immunology, histology, transcriptomic, metabolomic and SPECT-CT imaging) conducted to date, have not shown any significant adverse effects for the chosen regimen. Good Laboratory Practice toxicology studies were finalized and the submission to french health competent authorities of the proposal for modification of the dosing regimen for ITB was done.

The results of the work carried out in this project will ultimately provide health authorities with new operational solutions for the prevention of exposure to radioactive iodine, and thus contribute to the evolution of the "iodine doctrine".

Keywords: Potassium iodide, repeated prophylaxis, Thyroid gland, Thyroid hormones

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Renal Toxicity and Biokinetics Models after Repeated Uranium Instillation

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Uranium kidney toxicity is well known after acute uranium intake with a threshold around 3 µg/g of kidney. After chronical exposure kidney toxicity is controversial even after chronical ingestion of 600 mg/l during 9 months[1]. The main risk of contamination for workers exposed during uranium processing is inhalation and exposure situations can occur repeatedly.

Biokinetic model after uranium intake has been recently updated by ICRP[2] to include recent data. However most of experimental data either consider acute exposure or chronic ingestion through drinking water. To evaluate uranium intake at the kidney level after protracted exposure, assumptions are made but no experimental data are available to support them. The objective of this work is therefore to verify 1) if uranium biokinetic model developed from acute exposure are consistent with data obtained after repeated contamination of animals, 2) if renal toxicity modify uranium retention and excretion 3) if nephrotoxicity threshold can be predicted by the models?

Mice (C57BL6/J) have been exposed to different Uranium nitrate hexahydrate concentration (0.03-3 mg/kg/j) via intranasal instillation four times a week during two weeks. Uranium in urine and in tissue has been measured at several time points during exposure and up to 42 days post exposure. In addition the kidney toxicity has been measured, at the same time-point, in urines using Clusterin and Kim-1 kidney biomarkers and, confirmed by in situ kidney mRNA levels of early and late nephrotoxicity markers.

For the lower uranium concentrations, the experimental retention in kidney is well predicted by the usual biokinetic model when accounting for the exposure profile. However, whatever the concentration tested under this repeated exposure scenario, the amount of activity retained in kidney is lower than predicted by the models. Uranium biokinetics is also modified for higher doses probably due to tissue alteration and organ dysfunction not considered in biokinetic models. A clear nephrotoxicity is detected for the highest concentration 3mg/kg/j and suspected after a concentration of 1mg/kg/j.

Specified biokinetic models must be developed to take into account the possible modification of excretion and retention due to organ toxicity.

Keywords: Uranium, Biokinetic model, nephrotoxicity

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OS1.4 (T1.1-0554)

In vivo Assessment of Relative Genotoxicity and Tumorigenicity of Internal Low-dose Deta-radiation from Tritium

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Health effects of human exposure to internal beta-radiation from tritium are poorly defined (1). Prospects of the use of the thermonuclear fusion energy which is associated with production and handling of tritium, as well as current operation and expansion of fissile nuclear energy production, both make it imperative that biological effects of low-level tritium exposures are thoroughly studied. Addressing this issue, the Institute de Radioprotection et de Sureté Nucléaire has established a collaborative program with the Canadian Nuclear Laboratories to examine biological effects of tritium in mouse models in relation to matching external gamma-exposures (2). In a series of *in vivo* mouse studies, biokinetics of tritiated water (HTO) and organically bound tritium (OBT), molecular, cellular and tissues toxicity and genotoxicity in various organs, as well as tumorigenesis and lifespan effects were examined in several mouse models using a variety of chronic exposure lengths and concentration (0.01 to 3000 MBq/L). The effects were compared to those produced by external gamma-radiation at matching cumulative doses and dose-rates. Results suggest that: 1) OBT possesses higher biological effectiveness compared to HTO; similarly, HTO effects were in general more pronounced than effects of external gamma-radiation; 2) tritium exposures had a threshold of dose below which no effects were found; the threshold varied for different tissues and endpoints, and generally increased with the level of biological organization, i.e. molecular<tissue<organism; 3) as high as 500 MBq/L of HTO in drinking water produced no effect on lifespan in wild type CBA/CaJ mice when given for two weeks via drinking water; 4) high-doses of HTO (>1200 MBq/L or >0.5 Gy cumulative dose) produced a profile of neoplasia (>50 neoplasia types scored in 9 tissues) that was different from that produced by matching doses of gamma-radiation; 5) little or no effects of HTO (4.5 – 906 MBq/L) on tumorigenesis in mouse colon (APCmin/+) and breast (MMTVneu) cancer models were found; yet, a plethora of molecular, genetic and epigenetic changes were detected in these tissues. Overall, our results support the notion that internal beta-radiation from tritium may exert higher biological activity than external gamma-radiation; however a threshold mode of action was revealed since the effects were typically observed at concentrations that were substantially higher than current regulatory limits. Our data also provide essential knowledge for developing Adverse Outcome Pathways for tritium exposures.

Keywords: tritium, radiotoxicity, tumorigenesis

ACKNOWLEDGMENTS

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OS1.4 (T3.2-0546)

Evaluation the Risk of Secondary Malignancy Related to I-131 Therapy in Thyroid Cancer using Common Data Model

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There is controversy about the risk of secondary malignancy related to I-131 therapy. However, previous studies had been used limited medical information of the patients. Because they used data from limited information that are established for the special purpose such as payment, insurance, et al. The only way to overcome such limitation is using medical record of each hospitals. However, it is very laborious task to integrate and analyze multicenter data with entire information of patients. Recently, there is common data model (CDM) domestic and international consortium. We used CDM model to evaluate the risk of secondary malignancy related to l-131 therapy

Methods: The anominized data from common data model database of Seoul National University Bundang Hospital (SNUBH) was used this study. We selected the patient who had been undergone thyroid surgery in SNUBH from 2003-2018. After the selection of patients, we evaluated the diagnosis, blood test, et al before and after I-131 therapy. To correct the difference between the patients with and without 1-131 therapy, Propensity score matching method was used. The risk of secondary malignancy was evaluated using Cox hazard ratio analysis.

Results: From 2003 to 2018, total 5,351 thyroid cancer patients were enrolled. Among 5,351 thyroid cancer patients, 2,571 patients underwent I-131 therapy. After the propensity score matching, 873 patients were enrolled in each group. We compared the risk of secondary malignancy between two groups, and there no significant difference of the incidence of secondary malignancy (p-value, 0.82). The hazard ratio of second malignancy with I-131 therapy was 1.04.

Conclusion: From the analysis of DATA from CDM database of SNUBH, there was no significant difference of the hazard ratio between thyroid cancer with I-131 therapy and without I-131 therapy.

OS1.4 (T3.3-0586)

Repeated Potassium Iodide (KI) Treatment during Pregnancy Dysregulates Thyroid Function and Modulates Cortex Gene Expression of the Progeny

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Background: Nuclear power plant accidents can result in the release of large amounts of radioactive particles, in particular radioactive iodine (^{131}I). The WHO recommends a single administration of stable iodine in order to prevent the fixation of ^{131}I to the thyroid gland. In a situation of repetitive exposure to ^{131}I (*e.g.* Fukushima accident), additional doses of stable iodine may be necessary to ensure adequate thyroid protection. However, an excess of iodine for several days may induce adverse effects such as thyroid dysfunction (ex. hypothyroidism), especially in vulnerable population categories such as neonates and pregnant women.

Objective: The objective of this study is to evaluate the adverse effects, in a model of pregnant rats, of repeated administration of 1mg/kg/24h KI for 8 days (previously identified as an optimal treatment for thyroid protection in the adult model). A particular focus concerns the treatment period corresponding to the development of the thyroid and the central nervous system (CNS) of the progeny.

Methods: Pregnant rats were treated with either KI or saline water for 2 days (Group 1), 4 days (Group 2) or 8 days (Group 3). Clinical biochemistry, pituitary and thyroid hormone levels, as well as the expression of genes associated with thyroid and cerebral function of the progeny were analyzed 30 days post-weaning.

Results: Groups 1 and 2 showed hormonal imbalance, characterized by a significant increase of the triiodothyronine (T3) by 41 % and 17 %, respectively (p=0.01/p=0.03). In group 3, a significant decrease in the levels of both thyroid-stimulating hormone (TSH, −28%/p=0.05) and free thyroxine (T4, −7%/p=0.02) was observed. Repeated KI prophylaxis did not induce any change in gene expression in the thyroid for all groups. However, a significant increase in expression of Kcna1 (+26%/p=0.008), Mbp (+70%/p=0.03) and Mobp (+90%/p=0.03), known to be involved in synaptogenesis and myelination respectively, was observed in the cortex of animals from group 3, but not in group 1 and 2.

Conclusion: Our data show that repeated prophylaxis of KI over 2 or 4 days during gestation resulted in a T3-hyperthyroidism 30 days post-weaning. However, an 8-day prophylaxis induced an atypical T4-hypothyroidism combined with a decrease of TSH, which may reflect congenital hypothyroidism. These atypical hormonal changes were in line with variations of expression of genes implicated in brain plasticity. We thus conclude that repeated prophylaxis of KI over 8 days seems more deleterious than that of 2 or 4 days. Experiments are underway in order to challenge these preliminary results with CNS functional test (Rota-rod test, Forced Swim test and Y maze test). Our results are encouraging and will allow as to define and propose the most appropriate prophylactic duration for pregnant women.

Key words: potassium iodide, thyroid hormones, gestation, thyroid dysfunction, progeny.

OS1.5 (T1.1-0563)

Sfpi1 Gene Deletions Causing Acute Myeloid Leukemia Remain in the C3H Mice Spleen for a Long Time after X-irradiation

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Acute myeloid leukemia (AML) is one of the major cancers seen in human population exposed to ionizing radiation. In the several past studies using a model mouse of radiation-induced AML (rAML), hemizygous deletion of the *Sfpi1* gene (*HDSG*) has been recognized as an initiating event for developing rAML. We previously found that the cellular kinetics of hematopoietic stem cells (HSCs) with *HDSG* in γ-irradiated C3H mice with 3 Gy were involved in the process of rAML[1)]. Although the *HDSG* occurs in hematopoietic stem/progenitor cells (HSPCs) early following radiation exposure, the cellular kinetics of HSPCs carrying *HDSG* in various hematopoietic tissues after X-irradiation has been unclear. In this study, we examined the difference of frequency of *HDSG* in HSPCs between bone marrow and spleen of C3H mice irradiated with 3Gy of X-ray. Eight-weeks old male C3H mice were irradiated 3 Gy of whole-body X-ray (1Gy/min) and mice were sacrificed at 1, 4, 8, and 26 weeks after X-irradiation. Then, HSPCs were isolated from bone marrow of femur and spleen, the frequency of HSPCs with HDSG was analyzed by FISH method. The results showed that the number of HSPCs in bone marrow increased by X-irradiation within a week, and then gradually decreased to be lower than in the control group by 26 weeks. In the spleen, the number of HSPC increased gradually after X-irradiation. The frequency of HSPCs with *HDSG* in both bone marrow and spleen increased at a week after X-irradiation. The frequency of HSPCs with *HDSG* in bone marrow showed a gradual decrease from 4 to 26 weeks, whereas that frequency in spleen remained high even at 26 weeks after X-irradiation. Our study suggests that the HSPCs with *HDSG* migrated to spleen from bone marrow, and that HSPCs with *HDSG* continued to be present in spleen after X-irradiation. If major hematopoietic site could replace bone marrow with spleen following high-dose-rate X-irradiation, spleen might be one of the major site for rAML development.

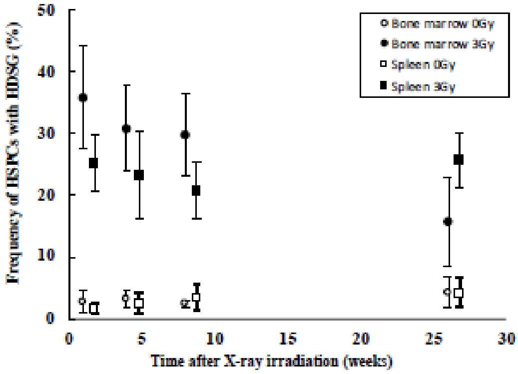

Fig. 1. The frequency of HSPCs with *Sfpi1/Pu.1* gene deletion in bone marrow and spleen

Keywords: radiation-induced *Acute myeloid leukemia (rAML), Sfpi1 gene, hematopoietic stem/progenitor cells*

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OS1.5 (T1.2-0176)

Lung Injuries after Stereotactic Radiation: Preclinical Modeling and Radiopathological Aspects

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Every year in France, 30,000 new cases of lung cancer are diagnosed, 85% of which are non-small cell lung cancers (NSCLC). Since 10 years, stereotactic body radiation therapy (SBRT) is proposed to inoperable patients as an alternative to surgery. This technique consists in irradiating a very small volume with submillimetric accuracy thanks to the convergence of mini-beams at the target center. This allows the delivery of ablative dose, from 6 to 20 Gy per fraction (e. g. 3x20 Gy). Globally, the main toxicity after SBRT is radiation pneumonitis (RP), which may progress, to radiation induced pulmonary fibrosis in the long term. Symptomatic RP incidence is from 10 to 20% of grade 3 or more, 2 years after the treatment[1]. The excellent tumor control obtained by SBRT (95% at 2 years[1]) increasingly encourages the use of this technique, despite the lack of hindsight on its toxicity. Here, the clinical practice precedes radiobiological knowledge. Currently, there is very little evidence of the response of small volumes of healthy lung tissue exposed to very high doses per fraction. Improving radiobiological knowledge on these points could provide a better understanding of mechanisms involved side effects development after SBRT.

The SARRP (Small Animal Radiation Research Platform) is a micro-irradiation system, image-guided and associated with treatment plan software that allows us to mimic SBRT in mice. Our aim is to acquire *in vivo* radiopathological data, considering volume, total dose and fractionation effect. The effect of the irradiation volume was studied thanks to 4 collimators: 10x10 mm^2, 7x7 mm^2, 3x3 mm^2 and 1 mm diameter, with a single dose of 90 Gy delivered in arc-therapy, centered on the left lung. Obliterans bronchiolitis compromised the survival of mice irradiated with 10x10 mm^2 and 7x7 mm^2 collimators. Irradiation with 3x3 mm^2 collimator induces early RP then the development of radiation induced pulmonary fibrosis restricted to the target volume. Because 3x3 mm^2 collimator allows us to study early and late effects of SBRT, we choose it for dose effect study, with a dose range from 20 to 120 Gy. Pulmonary volume irradiated in this configuration tolerates very high doses: mice survived until the end of the study, without any clinical symptoms. Lung parenchyma and bronchiolar epithelium react strongly from a dose of 60 Gy, with late impact on the contralateral lung. Necessary dose to trigger fibrosis is between 40 and 60 Gy. Fractionation of the dose was also studied, with 3x3 mm^2 collimator. Mice were irradiated with 4 different fractionation protocols, chosen to vary the BED_3 (Biological Effective Dose) and to keep a constant number of fractions to avoid changing the spread. Protocols were delivered in three fractions, on one week, with dose per fraction from 20 to 50 Gy in order to deliver BED_3 of 60, 108, 208 and 300 Gy near the target volume. Analyses are ongoing.

Finally, cellular response to high doses per fraction was determined, using human pulmonary cell lines irradiated with five fractionation protocols, with varying dose per fraction but constant BED_3, and analyzed 7 days post-irradiation. Custom card gathering 44 genes has been built and analyses are in progress.

This modelling provides *in vivo* radiobiological data, which combined with *in vitro* radiobiological data on different pulmonary cell lines, will provide a better understanding of the risks and effects of SBRT on healthy tissue.

Keywords: Radiation-induced lung injuries

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OS1.5 (T1.3-0338)

Feasibility Study on Deep Neural Network-Based Radiation Sensitivity Prediction Model using Gene Expression Data

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Purpose

In radiotherapy, the prediction of radiation sensitivity for various types of tumor has clinical importance. Researchers have mainly used the statistical methods to suggest the predictive model by analyzing the correlation between radiation sensitivity level represented by survival fraction at 2 Gy (SF2) values and gene expression data. In this study, we develop a deep neural network to model the nonlinear relationship between SF2 value and large amount of gene expression data.

Materials and Methods

The RNA sequencing gene expression profiling data based on National Cancer Institute (NCI)-60 cancer cell line was acquired from Cellminer (https://discover.nci.nih.gov/cellminer) database. We obtained SF2 values using clonogenic assay for 56 cells lines from previously published literatures. To ensure the reliability of the SF2 data, we used the SF2 values that has been reported in at least two different publications. We developed a regression model using deep neural network to estimate SF2 values from the gene expression data. To train the deep neural network model, leave-one-out cross validation approach has been applied. To validate the predicted SF2 values, Pearson's correlation analysis was performed using SPSS software (IBM SPSS Statistics version 25, New York, USA).

Results

As a result of the nonlinear regression analysis using 56 cell lines, the average relative error of the dataset was 2.82 ± 0.71%, and the median value of the relative errors was 1.37%. Data of 20, 28, 4, and 4 among the 56 test data were included in the groups with relative errors of less than 1%, 1 to 2%, 2% to 10% and 10% or more, respectively, which indicates that 85.71% of data have less than 2% of relative error. Data with relative errors above 10% had absolute SF2 measured values of 0.1 or less, and the absolute errors of these values with predicted ones were between 0.0072 to 0.0306. The Pearson correlation coefficient r value is 0.9988, indicating that the predicted SF2 values by our model and the measured SF2 values obtained from publications have a statistically strong positive correlation ($p < 0.001$).

Conclusions

Our results suggest that the utilization of deep neural networks is reasonably feasible to develop the prediction model for radiation sensitivity using large amount of gene expression data. This study is in early stage, and it is possible to expect various kind of applications in the future. In the future, we will be able to predict radiation sensitivity with more stable and higher accuracy by increasing the dimension of the data, utilizing better algorithms for identifying nonlinear relationships, and obtaining more accurate and larger amount of radiation sensitivity data.

Keywords: Radiation sensitivity prediction, Deep neural network, Nonlinear regression model

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OS1.5 (T4.2-0089)

Molecular Changes in Population Occupationally Exposed to Low -dose Ionizing Radiation: Interventional Radiology Unit Teams

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Interventional radiology team uses specific medical approach where procedures are usually done under lead aprons and in the vicinity of radiation beams. These are the main reasons for many occupationally related orthopedic conditions, as well as unwanted radiation-induced effects. Although the technological improvements decreased doses applied in such procedures, the number of them increased several fold. Strategic Research Agenda of the Multidisciplinary European Low Dose Initiative (MELODI) set priority research of cancer risks in occupational settings (particularly for doses ≤100 mSv) and individual radiation sensitivity. Therefore, the aim of the study was to assess cytogenetic changes in interventional radiology team and to relate them with occupational effective doses.
Exposed population (N=24) comprised of 18 male and 6 female volunteers, at the age of 40.96±9.73, BMI of 24.65±3.26 kg/m^2, and 29% of were smokers. They were recruited in three hospitals in Zagreb, Croatia. The control group (N=24) was matched for sex, age, BMI, and smoking status to reduce the possible confounders. Personal dosimetry using thermoluminiscent dosimeters showed average annual effective dose of 1.82±3.60 mSv (range 0 – 13.87 mSv). The baseline induction of primary DNA strand breaks was higher ($p<0.05$) in exposed population using the comet assay's tail length (Fig 1a). As for the abnormalities in chromosomal structure, the only significant increase observed was the frequency of nuclear buds (NBs) (Fig 1b), while other micronucleus test parameters did not differ among groups.
Based on our results, we managed to detect changes at the molecular level for the population occupationally exposed to low dose ionizing radiation. We also detected high inter-individual variability of the results. As for the personal dosimetry, more accurate protocols are needed in order to assess doses more precisely. Further studies are needed in order to improve occupational safety and to acknowledge ALARA principles.

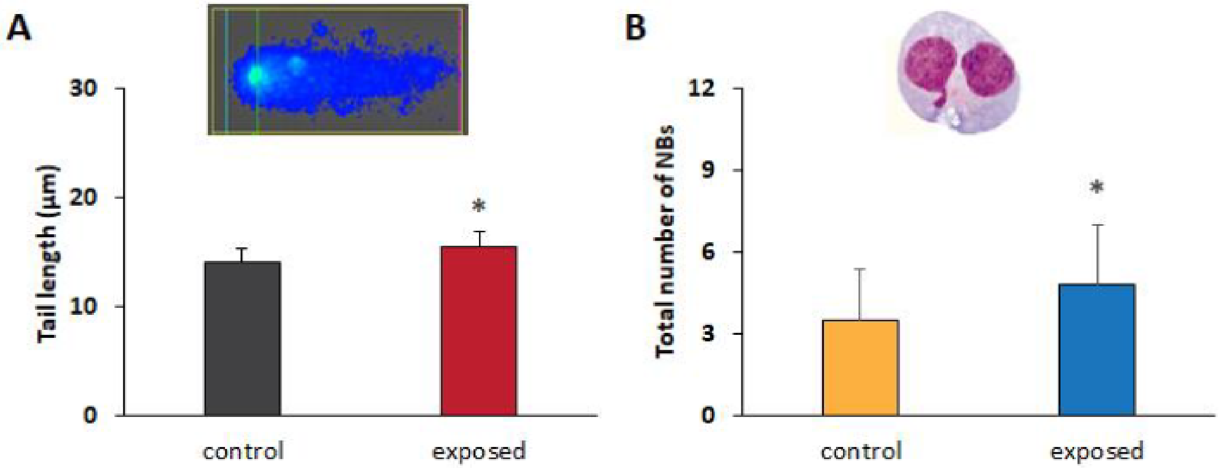

Fig. 1. Comparison of human biomonitoring biomarkers in interventional radiology unit workers (exposed) compared to control group. (A) the comet assay's tail length – the distance between green and magenta line; (B) total number of nuclear buds (NBs) – extruded nuclear material. Differences at $p < 0.05$ are considered significant (*).

Keywords: Human biomonitoring, Comet assay, Micronucleus test

OS1.5 (T4.3-0212)

Buccal Mucosa Cytogenetic Damage in Children Exposed to Diagnostic Sinus X-rays

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Ionizing radiation (IR) is a physical agent which interacts with cellular DNA molecules, producing a variety of primary lesions such as single- and double-stranded breaks, DNA–DNA and DNA–protein crosslinks that can result in gene mutation, gene amplification, chromosome rearrangements, cellular transformation as well as cell death. Hence, IR can induce chromosomal damage that can increase the risk of developmental abnormalities and cancer. Children exposed to IR are at a relatively greater cancer risk as they have more rapidly dividing cells than adults in addition to longer life expectancy. Therefore, we performed simultaneous physical dosimetry and buccal cell micronucleus cytome assay on pediatric patients before and after X-ray exam of sinus. The study involved 20 subjects (11 girls and 9 boys) aged 11.85 ± 3.60 years, and BMI < 25 kg/m^2. The samples of epithelial buccal cells were sampled before and 14 ± 1 days after the X-ray exam. Physical dosimetry was done using radiophotoluminiscent (RPL) glass dosimeters on four loci of the head. The buccal cell micronucleus cytome assay was done in order to evaluate DNA damaging, replicative, cytostatic, and cell death effects. Micronuclei as well as other biomarkers of DNA damage (nuclear buds and the so-called "broken eggs") and genomic instability (normal basal cells, normal differentiated cells, binucleated cells, cells with condensed chromatin, pyknotic cells, cells with karyorrhectic chromatin and karyolitic cells) were analyzed in a minimum of 2000/1000 cells per child, respectively. The observed doses were highest at the primary beam entry (370.8 – 1106.0 μGy) and were several fold higher than at the other loci. As for the biological parameters, we did not observe any significant DNA damaging effects. Yet, significant increase in cells with condensed chromatin was observed, indicating more cells undergoing early stages of apoptosis. It has to be pointed out that interindividual differences existed for each monitored child. Although we did not observe significant changes in majority of the parameters tested, some of the assessed parameters showed small increases in their values after the radiological examination indicating that further studies with larger samples are warranted. Based on our results, methods such as the buccal micronucleus cytome assay could be very useful in acute events where children are exposed to genotoxic agents especially from physical sources. Besides, this particular method could be used for monitoring genetic damage in children who are often exposed to diagnostic procedures, as it is a minimally invasive method of sample collection. In line with that, future research on biomonitoring and finding optimal biomarkers for low-dose IR exposure are required, especially in pediatric diagnostics to minimize the potential damage and maintain the optimal benefit for young patients. For now, we have to keep the annual absorbed dose as low as possible, especially at early ages, and conduct more research of the radio-sensitivity time window.

Keywords: Child population, X-ray dosimetry, Buccal micronucleus cytome assay

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OS1.6 (T1.1-0037)

Raman Spectroscopic and DFT Computational Study of the Interaction between Tritiated Water and λ-dsDNA

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To study the biological effects of HTO on internal radiation, it is necessary to clearly understand the damage of HTO in subcellular structural biomacromolecules and the distribution of this damage in biological space and time. DNA damage is caused by various intermediate stages in the biophysical and biochemical processes involved in the absorption and transport of HTO radiant energy, the excitation and ionization of molecules, the generation of free radicals, and the cleavage of chemical bonds. In this context, comparing DNA damage caused by HTO at different dose rates may be helpful in understanding the mechanism of radiation damage in HTO.

1×10^6Bq/ml HTO is mainly indirect effect of ionizing radiation during irradiation, which is ionizing radiation mediated by solvent molecules plays a major destructive effect on dissolved dsDNA. It can be seen from the experimental results that the increase in Raman characteristic peak intensity of bases and dRib occurring at a low dose (100-500mGy) of 1×10^6Bq/ml HTO can be attributed to base pair unstacking caused by various primary and secondary products of water radiolysis, and they have certain selectivity to DNA molecular structure, mainly destroying hydrogen bonds between bases cause base mismatches and modification in base structure, resulting in better unpairing of base pairs, ultimately leading to partial denaturation of dsDNA. The high radiation dose (4-8Gy) of 1×10^6Bq/ml HTO leads to a more pronounced decrease in Raman characteristic peak intensity. The Raman label band of the whole spectrum is split and broadened, indicating that the long-term action of free radical molecules is more likely to result in local modification in the sugar-phosphate backbone or nucleic base.

OS1.6 (T1.1-0037)

Raman Spectroscopic and DFT Computational Study of the Interaction between Tritiated Water and λ-dsDNA

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period of time is similar to γ-ray, mainly caused by direct ionization or protonation of the ray, which leads to the destruction of the base structure, especially the ring modes of the purine which causes the DNA skeleton to break. As the dose of HTO irradiation increases, leading the irradiation time increases, and the base pair unstacking effect also occurs.

The characteristic Raman peak of dRib exhibits the splitting of the Raman label band, the appearance of the new Raman peak and the dose-dependent disappearance of the characteristic line after high radiation dose of HTO (2-8Gy), indicating that the action of high dose HTO can cause the conformational transfer of the furanose ring or the cleavage of the covalent bond through tritium isotope effect, which can cause the single strand of the DNA backbone to break.

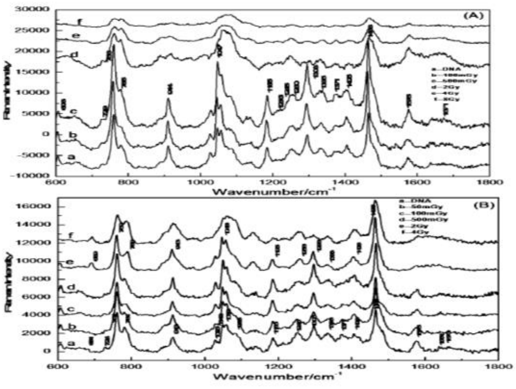
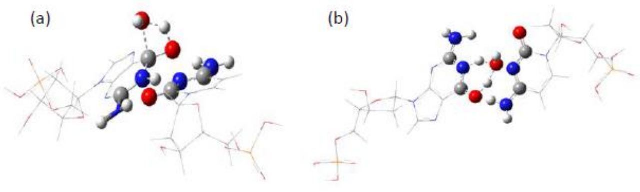


Fig. 1 Raman spectra of λ-dsDNA after 1.0×10^6 Bq/mL (A) and 1.0×10^8 Bq/mL (B) HTO irradiated at different doses. Background corrected spectra are presented in the region 600–1800 cm^{-1}. Peak positions of prominent Raman bands are labeled. The DNA concentration is approximately 20 mg/ml. For all measurements, the laser power at the sample space was 200 mW.

Six measurements of 120s were averaged for each spectrum.



ΔE=-9.67kJ/mol

ΔE=-145.55kJ/mol

Fig. 4 The obtained TSs of the cleavage of hydrogen bonding.

OS1.6 (T1.3-0245)

Low Dose Hyper Radio Sensitivity Effect of High Energy Neutron Radiation on *C.elegans*

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Previous studies about low Linear Energy Transfer(LET) radiation have shown that survival rate of cell lines decreases exponentially with the increase of radiation dose. In recent decades, scientists and physicists in medicine have found that some cell lines exhibit Hyper Radio Sensitivity(HRS) to low radiation doses that is not predicted by back-extrapolating the survival rate from higher doses[1,2,3]. However, the biological effects of low-dose for high LET radiation of neutrons need to be studied urgently because of the increasing possibility of human exposure to neutrons following the development of reactor technology, Boron Neutron Capture Therapy(BNCT) and so on. Thus, we studied the lifespan of *Caenorhabditis elegans* because of the better simulation of biological effects of individuals on environmental stress than cell lines. The results showed that all dose groups were significantly different with control group. With the increase of neutron dose, the lifespan showed an overall decreasing trend; especially the shortening effect of 1.83Gy on the lifespan. The result suggested that the HRS effect was firstly discovered of *C.elegans* from neutron radiation. As the dose is increased above about 6Gy, there is Increased Radio Resistance(IRR) until at doses beyond about 10Gy, radio resistance is maximal, and the life span of *C.elegans* follows the usual downward-bending curve with increasing dose. HRS effect of lifespan for *C.elegans* may result from the inability of low dose radiation to efficiently induce repair mechanisms, whereas higher doses cause enough damage to trigger repair responses. Although the molecular mechanism of HRS/IRR remains to be further elucidated, DNA damage and repair is believed to play an important role, especially in the damage and repair state of DSB. Therefore, failure or wrong repair of DSB caused by radiation becomes a key factor of cell death. Demonstration of HRS/IRR would provide mechanism of the phenomenon that so far observed in vitro and in animal models; on the other hand, if the possibility of HRS is too high in human normal tissues, this would argue against a therapeutic gain from ultrafractionation[4,5]. Nevertheless, below of 8.4Gy, the results of germ cell apoptosis for *C.elegans* were not shown significantly different with control, and the threshold for neutron radiation apoptosis is about 8.4Gy. The above results indicated that high energy neutron radiation might make a stronger damage effect at low dose, which provides scientific basis for neutron low-dose radiation protection.

Keywords: Neutron, HRS/IRR, Low dose

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OS1.6 (T1.5-0104)

The combined effects of exposure to low dose chronic radiation and parasite Crithidia bombi on bumblebee fitness

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The effect of radiation on wildlife is a key area of research, much of the current data come from laboratory experiments where study organisms were kept in optimum conditions. Effects have been measured in contaminated areas such as Chernobyl at much lower dose rates than predicted from laboratory data. One possible explanation as to why there is disconnect between the laboratory and the natural environment is multiple stressors.

We designed complementary field and laboratory studies to test whether a natural stressor (parasite infection) and increased radiation exposure interact to exacerbate the effects of radiation. Crithidia bombi is a common trypanosome parasite known to exhibit a negative effect when the bumblebees are under stress.

We conducted field sampling in Chernobyl and found that bumblebees from highly contaminated areas had a greater Crithidia bombi load and a lack of bees in the highest age class as judged from wing wear. Thus indicating that bumblebees from the most contaminated areas of Chernobyl have a higher number of parasites and reduced longevity.

We then tested this result under laboratory conditions. Bumblebees were infected bumblebees with Crithidia bombi and irradiated with dose rates encompassing those measured in the Chernobyl Exclusion Zone.

Our laboratory studied demonstrated that increased radiation exposure reduced the time taken for bumblebees to be infected with Crithidia bombi. The reduction in infection time was associated with increased parasite burden, which in turn, reduced individual bumblebee lifespan. Therefore, our laboratory data support field data from the Chernobyl Exclusion Zone.

We can conclude that low dose radiation exposure indirectly reduces longevity by exacerbating the effects of parasitism. This effect was smaller under laboratory conditions when compared to the field, which indicates that the optimal conditions and ab libitum food supply may have buffered against other stressors which would be encountered in the field.

This is the first truly complementary first study to combine laboratory and field studies in the field of radioecology. Our research significantly contributes to resolving the ongoing differences between laboratory and field studies which have measured the effects of radiation on wildlife. Furthermore, as the Bee is an International Commission on Radiological Protection (ICRP) Reference Animal, these new findings offer insight and provide much-needed information concerning the ICRP's Derived Consideration Reference Level.

***Keywords**: Radioecology, Multiple stressors, Bumblebees*

OS1.6 (T1.5-0660)

Protection of the Biota against Routine Radioactive Discharges around Nuclear Installations: MODARIA II – WG3

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The IAEA international project MODARIA II, run in the period 2016-2019, with the general objectives of enhance the capabilities of modelling the radionuclide transfer in the environment and assessing the exposure levels of the public and the environment in order to ensure an appropriate level of protection from the effects of ionizing radiation. Within this project seven working groups were formed, being the third of them: WG3, in charge of demonstrating the practical applicability of the radiological protection system to the biota in the particular problem of the normal operation discharges produced from the nuclear installations.

The main objective of the Working Group 3 is to check if the biota around nuclear installations is properly protected when the public was also protected. To investigate the applicability of the radiological protection of the biota in this particular case, the MODARIA II WG3has solved six problems:

1.- Existing national and international regulation has been compiled to define a practical method to carry out impact assessments on biota around nuclear installations.

2.- Computational codes used in different countries to perform dose assessments on the public has been improved to implement capabilities to assess doses to biota.

3.- Five scenarios of real nuclear installations worldwide have been defined and a number of integrated dose assessments on humans and biota were carried out on each of those scenarios.

OS1.6 (T1.5-0660)

Protection of the Biota against Routine Radioactive Discharges around Nuclear Installations: MODARIA II – WG3

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4.- The influence of including in the dose assessments a reduced number of Reference Animals and Plants, as recommended by the ICRP, or a real set of species existing around an installation, has been investigated.

5.- The influence of the size of the area, used to average activity concentrations to perform dose assessments on populations of animals and plants, was investigated.

6.- The problem of communicating to the stakeholders the dose assessments on biota were discussed.

All the achievements of this three years project are presented in this work.

Acknowledgements:

The main authors, respectively leader and scientific secretary of the group, present the work on behalf of all the participants in the WG3 of MODARIA II. In particular Valeria Amado, Amanda Anderson, Yuri Bonchuk, Stephanie Bush-Goddard, Peter Carny, Benoit Charrasse, Ari T.K. Ikonen, Elisabeth Leclerc, Justin Smith and Benjamin Zorko.

OS1.6 (T1.5-0663)

Why Many Chernobyl Radiation Effects Studies are Wrong

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Over the last three decades, many studies have evaluated the effects of radiation on wildlife within the abandoned area around the Chernobyl Nuclear Power Plant (ChNPP). This heterogeneously contaminated landscape provides a natural radioecological laboratory in which sampling sites can be identified across an activity concentration (and ambient dose rate) gradient. However, interpreting the results of Chernobyl radiation effects studies is challenging. Habitat variability, site connectivity, exposure history and proximity to areas of human activity can all influence study findings. Whilst researchers may have attempted to control for some of these factors, an implicit assumption in many previous studies is that ambient dose rate correlates with an organism's total absorbed dose rate. Drawing on data from our recent studies in the Chernobyl Exclusion Zone (CEZ; 2014 – 2018), this presentation demonstrates that the assumed correlation between ambient and total absorbed dose rate is a fallacy and that inferring dose effect relationships from ambient dose rates cannot be justified.

Within the CEZ, Cs-137, Sr-90 and actinides (Am-241 and Pu-isotopes) may all contribute to an organisms total absorbed dose rate from anthropogenic radionuclides. In May/June 2014, we sampled plant, animal and soil samples from a 0.4 km2 area on the edge of the 'Red Forest' approximately 5km west-southwest of the ChNPP. The activity concentrations of Cs-137, Sr-90 and actinides were determined and the results used to calculate internal and external radiation exposure. Our results demonstrated that: (i) the main contributors to dose are Cs-137 and Sr-90; (ii) the relative contribution of internal and external exposure to the total absorbed dose rate is highly variable; (iii) the ambient dose rate at the sampling site may underestimate the total absorbed dose rate by more than an order of magnitude.

In 2017 and 2018, we sampled small mammals from sites across an ambient dose gradient and quantified the total absorbed dose rate for each animal. The results clearly demonstrate that, even for sites outside of the 'Red Forest', the total absorbed dose rate for individual organisms is highly variable and that ambient dose rate cannot be used as a reliable proxy for total absorbed dose rate.

Our findings will inform future radiation effects research in Chernobyl and the derivation of regulatory benchmarks based on field dose-effect studies. These findings will also be important for regulators, providing a robust scientific basis for responding to stakeholders that use ambient dose rate-based radiation effects studies to challenge regulatory dose assessments.

Keywords: Chernobyl, dosimetry, wildlife

ACKNOWLEDGMENTS

We thank all those other scientists who have collaborated with us during the course of our studies. This work was largely funded under the TREE project (funded by NERC, the Environment Agency and Radioactive Waste Management Ltd.; https://tree.ceh.ac.uk) and a NERC Urgency Grant (Red Fire; https://www.ceh.ac.uk/redfire).

OS2.1 (T2.3-0196)

Machine learning for personal dose monitoring Insights into irradiation scenarios with a novel TL dosimeter

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Personal dose monitoring is one of the key concepts of a successful radiation protection program for occupationally exposed people. Thermoluminescence detectors are among the most widely used passive dosemeters. The individual monitoring service in the western part of Germany provided by the *Materialprüfungsamt NRW* will soon start to use thermoluminescence dosemeters instead of film dosemeters as well. The material used for the dosemeters is LiF:Mg, Ti (MT-N).
In routine dose monitoring, the luminescence light from a detector is used to estimate the received dose within the monitoring interval which is one month in most cases. From a radiation protection point of view, this retrospective dose estimation ensures rather a workplace safety compliance than an actual protection mechanism. Additional information about the irradiation scenario like the number of irradiation fractions or the irradiation date can be used to track the reason and circumstances of an exposure and can thereby improve an existing radiation protection concept by raising the chances to reduce subsequent exposures from the same source.

In this talk we will present results of our research on retrieving additional information about the irradiation scenario by analyzing the glow curves of thermoluminescence detectors using artificial neural networks. Artificial neural networks require large data sets to be trained before they can be used to predict the parameters of a new measurement. Therefore, more than 4000 glow curves have been recorded for several studies. Another option is the use of simulated glow curves. For this purpose, we have developed an empirical simulation approach to successfully augment our data set with glow curves from interpolated shape-determining parameters. A presentation of our simulation approach and the successful application on measurement data will be included in this talk.

Key results from our research are the estimation of the irradiation date in a single irradiation scenario with an accuracy of up to 4 days within a monthly monitoring interval and a simultaneous estimation of irradiation dose and date estimation with less than 10% uncertainty on the irradiation dose using simulated glow curves. Further results include the automatic computational separation of alpha and gamma or beta irradiation with up to 100% for identification of alpha irradiations and up to 80% for beta and gamma differentiation. Parts of the presented material have also been published in the *Radiation Measurements* journal [2,3].

Keywords: *Individual dose monitoring, thermoluminescence glow curve analysis, artificial neural networks*

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OS2.1 (T2.1-0300)

Experimental Evaluation of Scattered X-ray Spectra due to X-ray Therapeutic and Diagnosis Equipment for Eye Lens Dosimetry of Medical Staff

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The study presents an experimental evaluation of scattered X-ray spectra which medical staff is exposed during their procedure. Medical staff have been recognized as a highly exposed group due to low energetic X-rays, from the viewpoint of the eye lens dosimetry [1]. Additional monitoring of the equivalent dose to the lens of the eye should be implemented as required, in order to protect medical staff and comply with the annual dose limit, namely 20 mSv per year averaged for five consecutive years without exceeding 50 mSv in any single year.

We have been investigating the method to experimentally evaluate personal dose equivalent for the lens of the eye, $H_p(3)$ for medical staff. An appropriate air-kerma-to-personal-dose-equivalent conversion coefficient, $h_{pK}(3)$ should be determined, according to the actual situation of exposure of medical staff. The $h_{pK}(3)$ is strongly dependent on X-ray energy and one of the key parameters is the average X-ray energy which medical staff are to be received around their eyes. However, there is a few studies which experimentally investigated X-ray spectrum that medical staff are actually received during their procedures. In the study, the measurement of scattered X-ray from X-ray therapeutic and diagnosis equipment were conducted.

The CdZnTe semiconductor detector (Kromek GR-1) was employed for the measurement. Pulse height spectra from the detector was measured at the positions where a medical staff actually engage his/her procedure or treatment. A RANDO phantom and an ISO water slab phantom were chosen for simulating a medical staff and a patient, respectively. The CdZnTe detector was set the position close to the eyes of RANDO phantom and pulse height spectra were obtained at each position. The scattered X-ray spectra were then derived by de-convoluting the pulse height spectra using an unfolding code MAXED [2]. We present the measurement of scattered X-ray spectra at the position of medical staff of therapeutic diagnosis equipment such as Angiography, fluoroscopy and CT. We will also introduce the differences of scattered X-ray spectra obtained at the position of medical staff's trunk, neck and eyes.

Keywords: Eye lens dosimetry, Scattered X-ray, Medical staff

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This work was supported by the Ministry of Health, Labour, and Welfare, Japan.

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OS2.1 (T2.1-0665)

Alexandrite: Development of a Natural TL and OSL Radiation Detector

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The development of a functional and low-cost radiation detector is one of the goals of this investigation. Within this context, the potential of alexandrite ($BeAl_2O_4:Cr^{3+}$) as a dosimetric material was first investigated by thermoluminescence and optically stimulated luminescence measurements. Thermoluminescence (TL) is the light emitted by some materials upon heating after exposure to ionizing radiation, besides incandescence [1]. Optically stimulated luminescence (OSL) is similar to TL but in this case luminescence is stimulated by the absorption of optical energy instead of thermal energy [2]. OSL and TL have long been established as a reliable technique in dosimetry. Natural alexandrite as a TL/OSL detector was investigated because the chemical nature of this mineral, $BeAl_2O_4:Cr^{3+}$, that combines two binary oxides, BeO and Al_2O_3, thus naturally suggesting this mineral for dosimetric applications.

The pellets were produced using a Brazilian natural alexandrite sample. This natural sample was manually crushed, and the powder obtained was sieved by selecting grain sizes smaller than 0.35 mm. This powder was then mixed with an organic matrix based on a fluorinated polymer (for OSL measurements) and special high temperature glue (for TL measurements). Irradiation was executed at room temperature using the built-in $^{90}Sr/^{90}Y$ beta source of the TL/OSL reader delivering a dose rate of 10 mGy/s. TL glow curves were obtained using a heating rate of 1 K/s, from RT to 500 K. The activation energy (E) and frequency factor (s) of the TL glow peaks were obtained by glow curve fitting using the GlowFit software [3], assuming a first-order kinetics. The OSL signal was stimulated in CW mode using blue LEDs (470 nm, FWHM = 20 nm) delivering a maximum power density of 80 mW/cm^2 of at the sample position. The TL and OSL measurements were taken in terms of dose-response (beta dose from 0.1 to 5 Gy), repeatability, reproducibility, and fading. In addition, optical absorption measurements were carried out in the range from 200 to 700 nm on the pellets before and after 10 Gy beta dose.

Alexandrite pellets showed sensitivity in a wide range of doses, especially low doses (less than 1Gy), which are often used in medicine area that uses ionizing radiation. In addition, the results showed that the TL and OSL integrated signal varied linearly with the irradiation dose. There is no relevant effect of radiation damage by the optical absorption spectrum.

Keywords: Dosimetry, Alexandrite, Mineral.

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OS2.1 (T2.1-0666)

Luminescence of UV-irradiated α-Al2O3:C,Mg

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Currently, there is an increased use of artificial UV sources and overexposure is the main factor that causes skin cancers, erythema, opacification of the lens and inflammation of the eyes. Due to this a search for new materials and practical methods for UV detection and dosimetry becomes important. Al_2O_3:C,Mg is a very sensitive luminescent material for applications in optical data storage and imaging. It has been successfully used in dosimetry of neutrons, protons and heavy charged particles. Although Al_2O_3:C,Mg is very promising in the monitoring of beta, gamma e X radiation, there are no specific previous studies on the influence of ultraviolet (UV) radiation on this crystal nor about its possibility of use as an UV-dosimeter. In this work, Al_2O_3:C,Mg single crystal was investigated as a potential UV detector, using the thermoluminescence (TL) and optically stimulated luminescence (OSL) technique. Thermoluminescence (TL) is the light emitted by some materials upon heating after exposure to ionizing radiation, besides incandescence [1]. Optically stimulated luminescence (OSL) is similar to TL but in this case luminescence is stimulated by the absorption of optical energy instead of thermal energy [2]. OSL and TL have long been established as a reliable technique in dosimetry. UV irradiation was performed with a Hg lamp (2.44 W/m^2 at the sample position) with different illumination times (10 to 60 s). TL and OSL measurements were carried out using a Risø TL/OSL reader (model DA-20). TL glow curves were obtained using a heating rate of 1 K/s, from RT to 500 K. The activation energy (E) and frequency factor (s) of the TL glow peaks were obtained by glow curve fitting using the GlowFit software [3], assuming a first-order kinetics. In addition, the OSL emission was stimulated using blue light emitting diodes (470 nm, FWHM = 20 nm) delivering 80 mW/cm^2 at the sample position in CW mode. TL results showed three low intensity peaks at low temperatures (about 320, 350 and 375 K) and the main peak at 455 K. The results of the OSL were analyzed by the initial intensity and of the curve, showing a typical decay curve, even for a short exposure to UV. We concluded that TL and OSL signals of Al_2O_3:C,Mg are promoted by UV radiation. In addition, for both TL (area of the main peak) and OSL (area under the curve) results, UV irradiation led to a non-linear response as a function of the irradiation exposure that could be approximated by a linear behavior for short irradiation times (< 15 s), while showing a trend for saturation for longer irradiation times. Saturation was attributed to the unique characteristics of the absorption and dissipation of the incoming radiation, as well as changes in the F-type/F-aggregate concentrations.

Keywords: *UV Dosimetry, Al_2O_3:C,Mg, Luminescence.*

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OS2.1 (T2.2-0295)

IDAC-Bio, an internal dosimetry software based on the new ICRP biokinetic models and specific absorbed fractions

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It is known that ionizing radiation has health effects on humans. Therefore, it is an evident need to assess radiation absorbed dose to various population groups for miscellaneous exposure situations – including planned, existing and emergency situations – with appropriate accuracy. International Commission on Radiological Protection (ICRP) has developed a new computational framework for internal dose estimations, summarized for more detailed and physiologically correct biokinetic and anatomical models than before. Therefore, the ICRP is currently producing new dose coefficients for occupational intakes of radionuclides (OIR) and environmental intakes of radionuclides (EIR) – series, which supersede the earlier dose coefficients. However, the ICRP only published bioassay information (retained and excreted activity) for single acute intakes of radionuclides integrated over 50 years.

IDAC-Bio is created in Matlab with a user-friendly graphical interface and compiled into a Standalone executable file containing all code and data. The software follows the ICRP computational framework for internal dose. The software uses the raw data of the biokinetic models and numerically solves the biokinetic model and calculates the absorbed doses to the reference phantoms. This allows the software to calculate absorbed dose for the nuclear decay data in ICRP publication 107. By performing absorbed dose and committed effective dose calculation from the raw data there is also a possibility to select an arbitrary integration time or different inhalation exposures scenarios, e.g. a continuous intake 9 hours a day. IDAC-Bio, is a further development of the internal dosimetry program IDAC-Dose2.1 and allows retrospective absorbed dose calculations. In the software parameters such as data from real measurements, i.e. time, radionuclide activity and organs are selected. IDAC-Bio uses the ICRP biokinetic models and fit the measured activity to these models to estimate the realistic exposure scenario. From the estimated exposure scenarios then the absorbed dose and committed effective are calculated. This retrospective calculation will allow a more realistic absorbed dose estimation for persons investigated for internal contamination than possible up to now.

The software was tested and validated against the published ICRP dose coefficients. The retrospective dose calculations were tested on a power plant worker, who had undergone several whole-body measurements. IDAC-Bio successfully manage to create a good fit, resulting in a realistic time varying activity concentration for the worker. Fitting the ICRP models to the measured data points creates a possibility to perform realistic absorbed and effective dose estimations for arbitrary exposure scenarios.

Keywords: *ICRP, Internal dosimetry calculations, IDAC-Bio*

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OS2.1 (T2.4-0483)

Inter-laboratory comparison of TL/OSL method on mobile phone glasses in retrospective dosimetry group in South Korea

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In a radiation emergency, several techniques for dose reconstructions of individuals have been developed. The methods vary from analysis for dicentric chromosomes in blood samples[1)], an electron paramagnetic resonance (EPR) for tooth samples[2)], thermoluminescence (TL) / optically stimulated luminescence (OSL) measurements for components in a mobile phone[3, 4)], Monte Carlo calculations with exposure scenarios[5)], etc. Since the TL/OSL method provides faster dose reconstruction and relatively lower detection limit[3, 4)], it is suitable for earlier retrospective dosimetry than other methods. Korea Atomic Energy Research Institute (KAERI) and Korea Institute of Nuclear Safety (KINS) have been cooperated to develop and apply TL/OSL method using mobile phones in retrospective dosimetry group in South Korea. Most of radiation accidents reported by KINS were occurred in working places of non-destructive testing using high activity radiation sources. Therefore, validation of the TL/OSL method by inter-laboratory comparison was considered with a virtual accident in radiation testing room of real work places. The concept of the accident is an exposure of a radiation worker with a certain distance from an unshielded Ir-192 source. Several mobile phones will be located on the chest, hip and side, which are common places of mobile phones, of a physical human phantom. Materials for the measurement will be bottom display glasses, resistors, and SIM cards. Sophisticated protocols for glasses (TL), resistors and SIM cards (OSL) will be applied. As a reference dosimeter, TLD (LiF:Mg,Cu,Si) will be used. The TLDs will be positioned on the mobile phones and in and outside of the phantom to estimate surface and organ doses as well as phone doses Computational dose calculations of mobile phones, anthropomorphic phantom and TLDs will be carried out using GEANT4 (GEometry ANd Tracking 4) code with mesh-type reference computational phantom and specially designed mobile phone phantom.

Keywords: *Emergency dosimetry, Inter-laboratory comparison, TL, OSL, mobile phone glass*

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OS2.1 (T2.1-0079)

Optimization of Luminescent Dosimeter Energy Response by Geometry Modification

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Objective: The purpose of this study is to establish the effect of dosimeter geometry on the luminescent dosimeter energy response for different radiation beams and optimization of this response.

Methods and Materials: In this work, we calculated by Monte Carlo simulation, using MCNP code, the energy response of three kinds of non-commercialized luminescent dosimeters: a) RPLGD (FD-7 glass), b) TLD (LiF: Mg, Ti) and c) OSLD (Al_2O_3: C). The three dosimeters present cylindrical geometry of 2 mm of thickness and different diameters ranging from 1 to 13 mm. The used radiation beams are 70 kV X-ray beam, clinical photons beam of 15 MV and clinical electrons beam of 9 MeV.

Results: Our results showed that the dosimeters energy response increases from 0.88 to 1.02 for clinical beams of photons and electrons. For the kV X-ray beam, a reduction in energy response of 8.8%, 1.4% and 8.7% is reported when diameter varies from 1 to 13 mm for RPLGD, TLD and OSLD, respectively.

Conclusion: In conclusion, this study confirms the non-neglected effect of the dosimeter geometry on the energy response for kV X-ray beam and the clinical photon and electron beams. Further study is proposed according to the luminescent dosimeters application (RPLGD, TLD and OSLD) for the routine quality assurance (QA) check for clinical electrons beam.

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OS2.1 (T2.3-0481)

Radiation Imaging Technique Based on Rotational Modulation Collimator (RMC)

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Environmental monitoring of radioactivity can be greatly assisted by the radiation imaging technique, which detects and visualizes ionizing radiations from the radioactive material. Gamma-ray and neutron imaging, in particular, plays a key role for the environmental safety of nuclear techniques in various applications as well as the homeland security and safeguards. For the development of a radiation imager capable of visualizing both neutrons and gammas, the rotational modulation collimator (RMC)-based imaging technique was attempted as an economical and effective approach which offers less complexity in the system configuration [1,2]. Recently, we have suggested an RMC system combined with the pulse shape discrimination capable Cs_2LiYCl_6:Ce (CLYC) scintillation detector as a dual-particle imager. Batches of Monte Carlo simulations were performed using Monte Carlo N-Particle version 6.1 (MCNP6.1) code to investigate the feasibility of developing CLYC-based RMC system, and an image reconstruction algorithm implementing maximum-likelihood expectation-maximization (MLEM) approach was developed. We also optimized the collimator mask design parameters for the dual-particle imaging purpose and proposed a new asymmetric mask design to remove the ambiguous artifact without compromising the imaging capability [3]. In this paper, we introduce the newly-developed dual-particle imager and report results from the measurement experiments conducted for verifying dual-particle imaging capability at a mid-range field using gamma-ray and neutron sources. We report recent updates on the results from the CLYC-based RMC examined for a single source and multiple sources imaging and evaluate the imager's performance in a mixed neutron/gamma-ray environment to investigate the repeatability and the reproducibility of the result by conducting additional experiments. The signal-to-noise ratio and the structural similarity index was calculated to evaluate reconstructed images of the source distributions quantitatively. Measurement results showed that the modulation patterns obtained from the RMC imager showed good repeatability and reproducibility and well matched MCNP6.1 simulation results. Reconstructed images was shown to make good estimations on the radiation source location. In this study, it was shown that both gamma rays and neutrons can be visualized to estimate the source location of radioactive materials utilizing a CLYC-based RMC system. The experiments gave promising results on utilizing CLYC-based RMC for radioactive material monitoring and detection.

Keywords: *Dual-particle imager, Rotational Modulation Collimator (RMC), Radiation Imaging*

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OS2.1 (T2.1-0628)

Introduction of Mobile Dosimetry Laboratory in KHNP

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Dose assessment for emergency workers was delayed due to the collapse of the dose evaluation system at the site during the Fukushima nuclear accident. Prior to the accident, only fixed whole body counters (WBCs) were fitted, so the role could not be performed by the tsunami. Since then, the Fukushima NPP currently has more than 20 mobile WBCs. KHNP has been operating stand-type and bed-type WBC using NaI and HpGe, respectively. Compared to pre-accident Fukushima NPP, it maintains a good operating system in terms of analysis of radionuclide and shielding design. However, it has been operated on a fixed basis within NPP. When WBC is mounted in the vehicle for mobile operation, low center of gravity design and attachment of auxiliary equipment should be considered. Shielding is also important to be able to assess the dose of workers in a high background to a low level below 0.1mSv. Based on comparison between the existing WBC at the NPP and lightweight WBC, it was deemed most efficient for mobile WBCs to follow the shielding type used by the existing WBC in consideration of their operation in the event of an accident. We chose a trailer that could only be connected to the vehicle if needed, rather than a truck that was difficult to manage. A scintillator detector with a high measurement efficiency and a shadow shield method to minimize the impact of a high-level background were adopted. Additional radiation portal monitor was required at the hatch to filter out the contamination of the subjects and protect the equipment, the radiation zone monitor was added for identification of residential requirements, and automatic correction of measurement signals and air conditioning facilities were considered to ensure safety due to changes in the instrumentation environment. Considering the identification of contamination and security issues, the vehicle's location and surrounding radiation dose rates while moving to the accident site were transferred to the accident control department in real time, and additional systems were constructed that can share the employee's measurement screen and dose record. Space for in-vivo ESR was provided for external dose evaluation, and in-vivo ESR can be detached so that beds for patients can be placed if internal dose assessment system is operated only. Although the U.S., Japan, France and others are building up their operational experience by introducing mobile dose assessment systems ahead of Korea, it has not been long since Korea introduced them in only a few places. Future procedures need to be established and accumulated operational experience is necessary to improve the system.

Fig. 1. Mobile Dosimetry Laboratory in KHNP

Keywords: *Nuclear power plants, Whole-body counting*

OS2.2 (T2.1-0051)

Proposed Guideline for Dose Assessment after Exposure to I-131 in an Accidental Situation

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The events in which individuals can become internally contaminated with I-131 may cover a wide range of possible situations, ranging from an accidental exposure in a medical, industrial or research environment that involves only a few people to an accident at a nuclear power plant or a radiological terrorist event, with dozens of potentially contaminated individuals.

Dose evaluations of these individuals may include the analysis of bioassays: such as the measurement of thyroid, urine and/or nasal samples. These measurements may support medical decisions, which may be based on the projected thyroid dose and / or the effective dose. The purpose of dose assessment in these situations is to provide objective information that contributes to make decisions about follow-up actions after an incident, to comply with legal regulations and to improve conditions in the workplace.

In this work, it is proposed a nine-step guide for the assessment of exposure to I-131 after real or suspected abnormal events or in the case of a positive result during triage or routine monitoring. This methodology allows giving recommendations on the type, number and time at which measurements should be made. This guide provides an effective handling of I-131 accidental exposures, contributing in the support for decision-making on follow-up actions (e.g., thyroid blockade with KI), as well as it allows verifying compliance with the legal regulations.

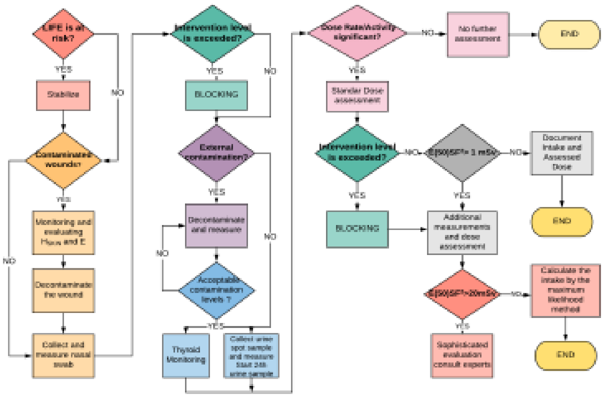

Fig. 1. Guide for the assessment after exposure to I-131

Keywords: I-131 accidental exposure, thyroid dose, dose assessment

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Oral Session T2

OS2.2 (T2.2-0297)

IDAC-Radon, an Internal Dosimetry Code for Radon and its Progenies using the New ICRP Biokinetic Models and Specific Absorbed Fraction Data

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The US environmental protection agency (EPA) estimates that nearly one in 15 homes in the United States has a radon level that should be reduced. The Swedish Public Health Authority estimates that the number of homes in Sweden with radon levels above the guideline value of 200 Bq/m^3 is at least 400,000, which means one in twelve homes. After smoking inhalation of radon (Rn-222) is the second leading cause of lung cancer. Radon forms when uranium in water, rocks, and soil begins to break down, releasing radon gas into the ground beneath buildings. Radon can enter through e.g. cracks in foundation walls and floors, or fireplaces and furnaces. International Commission on Radiological Protection (ICRP) have in ICRP Publication 137 revised both their biokinetic models and absorbed dose calculations for acute intake of radon. The ICRP dose coefficients includes progenies of the radon once the radon is inhaled and inside the body. However, the progenies will also be inhaled as these also occur in the air. The progenies will have different air flow rates leading to different activity concentration than radon once they are inhaled.

IDAC-Radon is a software code dedicated for internal radon dose calculations. The software performs absorbed dose calculations based on the full biokinetic transfer, and including inhalation of progenies. As the biokinetic calculations are solved numerically, all parameters and data connected to the radon calculations can be built-in to a complex biokinetic system. This allows for continuous inhalation at different breathing rates e.g. continuous inhalation 9 hours a day, 5 days a week. The users select which of the radionuclides to perform estimations for. Add the activity concentration, the integration period, the type of continuous inhalation and the inflow and outflow transfer rates in air for the different progenies. Parameters which all affects the number of radioactive particles inhaled. IDAC-Radon uses the biokinetic compartment models of ICRP Publication 137, the nuclear decay data of ICRP Publication 107 and the specific absorbed fraction data of ICRP Publication 133 to calculate organ absorbed doses and the effective dose defined in ICRP Publication 103.

IDAC-Radon uses a fast differential equation solver to calculate the activity distribution and performs absorbed dose calculations directly on the result. This enables the possibility to set arbitrary integrating periods, modify, pause or stop the inhalation periods during the stepwise calculation procedure. IDAC-Radon is a code created to facilitate more realistic absorbed dose calculations from exposure of radon.

Keywords: Radon, ICRP, Internal dosimetry calculations

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OS2.2 (T2.2-0310)

Development of Specific Absorbed Fractions for Reference, Age-dependent Members of the Public

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The computation of dose coefficients due to intakes of radionuclides requires knowledge of the fraction of an emitted radiation's energy from a source region which will be absorbed in a target tissue. This quantity is called an absorbed fraction and when divided by the mass of the target tissue becomes a specific absorbed fraction. Biological variation makes calculation of person-specific absorbed fractions impractical. Instead, for radiation protection purposes, a set of reference individuals is defined and specific absorbed fractions are tabulated for those reference individuals. ICRP Publication 133 contains specific absorbed fractions for the reference adult. The adult values were used to support the computation of occupational dose coefficients found in ICRP's Occupational Intakes of Radionuclides series. In the present work, we describe the development of specific absorbed fractions for a series of age-dependent reference individuals. This development can be summarized in three steps: (1) definition of the reference individuals' geometries with particular emphasis on source and target region masses, (2) radiation transport simulations in voxel and stylized models representing such geometries, and (3) quality control checks for consistency on large sets of data. The result is sets of specific absorbed fractions for 6 reference ages (newborn, 1-, 5-, 10-, 15-year old, and adult), 2 genders, and 4 radiation sources (alpha, electron, photon, and neutron.)

OS2.2 (T2.1-0970)

Implementation of a Triage Monitoring Program for Short-lived Radionuclides in Israel

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Monitoring of occupational radiation exposure is required according to the International Basic Safety Standards (BSS)[1] to verify the compliance with the requirement for protection and safety of occupationally exposed workers. When it comes to monitoring of internal exposure of short-lived radionuclides, periodic monitoring techniques conducted quarterly or even monthly, such as Whole-Body Counter (WBC) and radio-toxicology measurements of urine samples, will not be effective to detect internal exposure that occurred shortly after the previous measurement. The Swiss Personal Dosimetry Ordinance include a screening measurement (triage monitoring) conducted in the workplace to identify workers suspected of internal exposure[2]. First, an in-vivo screening determines whether there is a suspicion an internal exposure had occurred and if so, a WBC measurement or in most common scenarios the measurement of a urine sample is conducted for the quantification of the dose. This in-vivo monitoring program was also recommended in other international regulations, ISO standard and recommendations[3-5].

According to the Israeli Safety at Work Regulations[6], a worker suspected of internal exposure is monitored quarterly. This requirement has been fulfilled by measuring the worker in a WBC or by conducting a radio-toxicology urine sample measurement. Our aim was to study the feasibility of the implementation of the Swiss procedures and above-mentioned international standards and recommendations[2-5] in Israel taking into account the Israeli work culture. Preliminary work included a comparison between four common radiation detectors in Israel and their compatibility for the in-vivo screening of ^{131}I and ^{99m}Tc[7]. The screening protocol designed according to the international standards: workers working with ^{131}I were screened weekly and workers working with ^{99m}Tc were screened daily. If the threshold is exceeded, a radio-toxicology urine sample was sent to the National Internal Dosimetry Laboratory at Soreq NRC.

The study was conducted in a major hospital, a regional hospital and the largest manufacturer of radiopharmaceuticals in Israel. Some major challenges arouse in the implementation of the planned procedures. The main finding of the study was the inconsistent reporting and lack of workers' understanding of the method and importance of immediate measurement of a urine sample. Urine samples were not sent in real-time to the dosimetry lab. In some cases, external contamination has been reported as the screening result and a second measurement was not conducted after the removal of the contamination. Another finding was the use of the radiation monitor itself; part of the staff didn't understand how to operate the monitor, leading to wrong use of units and failure to send urine samples in possible intake situations.

For the successful implementation of the triage monitoring a few modifications have been made: additional training of all workers on the measuring technique and a change in the time of the self-measurement (mid-shift as opposed to the end of the working day). In addition, a surface contamination measurement has been added, in order to comply with the ISO standard[5]. The results show a promising outcome, that allow for a future implementation of the triage monitoring as part of the occupational dose monitoring regulations in Israel.

Keywords: Occupational dose monitoring, Short-lived radionuclides, Triage monitoring

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Oral Session T2

OS2.2 (T2.1-0280)

Development of Integrated in Vivo Monitoring Technology and System

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In vivo monitoring technology, when its detection limit meets the requirements, is prioritized by IAEA for internal radioactive contamination monitoring due to its high speed and accuracy. In vivo monitoring involves a wide range of aspects like gamma-ray detection and analysis, calibration phantom, virtual calibration, wound measurement, dose assessment and so on. Integrating these parts and form a complete all-in-one monitoring system will greatly meet practical needs and be of application value. However, because of its technical depth and complexity, the realization of this idea is a big challenge.For these reasons, China Institute for Radiation Protection has carried out a series of in-depth researches, and have finally established a complete system of in vivo monitoring technology. The details of this system are to be briefly provided in the following part of this abstract.

In terms of calibration technology, Chinese torso physical phantom CIRP-Torso for lung counter, Chinese aged BOMAB phantom CIRP-BOMAB-Age (including 6 age groups), Chinese reference BOMAB phantom CIRP-BCAM and sBCAM(in solid) for WBC are developed, and phantom fabrication based on 3D printing technology is preliminarily explored. In terms of dynamic virtual calibration, a sophisticated method is established based on the Chinese reference voxel and mesh phantom. And in aspect of dose assessment, INDOSE, a software with the ability to solve biokinetic equation, and calculate organ equivalent dose and effective dose, is developed. In addition, wound contamination depth measurement method is established based on computational phantom and Monte Carlo simulation.In particular, in the part of gamma spectrum analysis, a set of exclusive algorithms designed for vivo monitoring has been developed, and on the basis of it, a special gamma spectrum analysis program package called Gamma+ has been created. Compared with Genie 2000, Gamma+ performs better in the analysis of overlapping peaks, peaks with high continuum and in low energy region. Meanwhile, in view of fixed measuring position in in-vivo measurement, we came up with a new method to reconstruct the emitting spectra by nuclides base on sparse signal principle. Moreover, fast anomaly detection technology based on machine learning has been studied recently in order to realize fast measurement.

Based on this technology system, several applicable all-in-one in vivo measurement systems have been developed. They include: a standing large-volume NaI type whole body counter, equipped with Simple In Vivo-S which is an intergrated software system with the ability to conduct data acquisition, spectrum analysis, virtual calibration, dose assessment and personnel data management; a portable thyroid counter with integrated software Simple In Vivo-T; and an in-vehicle monitoring system with Simple In Vivo-V.These research results on integrated technology provide a high-precision in vivo monitoring hardware and software system for application, and offer innovative ideas for the future development of in vivo monitoring technology..

Keywords: *in vivo monitoring, integrated technology, measuring system*

OS2.3 (T2.2-0669)

Do We Need To Be Concerned With (α,n) Neutrons in Internal Dosimetry

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International Commission on Radiological Protection (ICRP) Publication 133 provides specific absorbed fractions (SAFs) for fission-spectrum neutrons that are associated with radionuclides that can decay by spontaneous fission. This is in keeping with the ICRP formalism that SAFs are only produced for radiations emitted directly in the decay of radionuclides; however, alpha-emitting radionuclides which may potentially lead to higher levels of neutron doses outside the source organ due to the generation of (α,n) neutron. The generation of (α,n) neutron sources can occur in any radionuclides by the emission of alpha particles of sufficient energy to meet the threshold for that reaction. According to ICRP formalism, the dose due to neutrons from (α,n) reactions would technically be included in the SAF value for the alpha emission.

As a first level attempt at assessing the order of magnitude of internal dose associated with (α,n) neutron emissions, computations were carried out for Am-241, Pu-238, Pu-239 and Pu-240 placed in materials with elemental compositions corresponding to bone, liver and lung. These computations were performed with SOURCES-4C for each composition. In the resulting values reported in Table 1, only the activity of the parent radionuclide was considered and not their progeny. The number of neutrons emitted per cm^3 of the source organ/tissue were computed with the code assuming a 1 Bq/cm^3 activity concentration. This table indicates up to 14-15 times more (α,n) neutrons than spontaneous fission neutrons could be emitted from these organs and in contrast Pu-240, which has a "large" spontaneous fission emission probability, produces about 0.3-0.4 (α,n) neutrons per spontaneous fission neutron. Based on this ongoing work, the presentation will discuss whether this phenomenon is an interesting physics study or whether it needs to be included in internal dosimetry studies. The presentation will include data for a larger number of alpha emitters, including spontaneous fission radionuclides. In addition the magnitude of the impact of including (α,n) neutrons in the SAFs for alpha emission will be presented based on anthropomorphic phantom dose calculations.

Table 1. Comparison of (α,n) neutrons to spontaneous neutron source terms in bone, liver and lung compositions.

Nuclide	Neutron Source (n/cm^3-sec per 1 Bq/cm^3)					
	Bone		Liver		Lung	
	(α,n)	SF	(α,n)	SF	(α,n)	SF
Am-241	5.5E-08	9.8E-12	5.9E-08	9.8E-12	5.8E-08	9.8E-12
Pu-238	5.6E-08	4.1E-09	6.0E-08	4.1E-09	5.9E-08	4.1E-09
Pu-239	4.1E-08	6.5E-12	4.5E-08	6.5E-12	4.4E-08	6.5E-12
Pu-240	4.1E-08	1.2E-07	4.5E-08	1.2E-07	4.5E-08	1.2E-07

Keywords: (α,n) neutrons, *spontaneous fission, specific absorbed fractions*

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OS2.3 (T2.1-0375)

Internal Dose Assessment in Occupational Unexpected Exposure to Xe-133 and Xe-135

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Following an unexpected exposure to high concentration of noble gases in which a reactor's worker was immersed during some minutes, internal dose assessment was required although the personal dosimeter's worker readings were not significant. This requirement arose as a result of having obtained positive values of activity of Xe-133 and Xe-135 in the thorax measurements performed on the worker from a few hours to several days after the event. Because the dose estimation methodology for these noble gases considers only external exposure [1, 2], the requirement of internal dose evaluation presented a challenge to perform it without biokinetic models or dose coefficients of reference for occupational intake of these isotopes, in addition to the need for a rapid result of the dose estimation.

The intakes of the xenon isotopes were estimated from the reconstructed concentration of gases at the time of the event, which were 130.2 MBq of Xe-133 and 10.22 MBq of Xe-135. Using the thorax measurements and the estimated intakes, it was observed that the measurements of internally deposited xenon were reasonably consistent with the behavior of retained xenon in systemic tissues, mainly in fat, observed in subjects and in model predictions found in literature [3, 4, 5, 6]. The Xe-133 dose factors recommended by ICRP 128 for patients treated with this radionuclide by re-inhalation of the gas for 10 minutes [6] were considered for this scenario. For taking into account the dose contribution for Xe-135 inhalation, MIRD methodology was implemented. Additionally, positive results of HTO and I-131 in urine bioassays was evaluated using the metabolic models recommended for the worker, under the assumption of acute inhalation and no contribution from previous intakes of these radionuclides. It was estimated a value of the committed effective dose for inhalation of Xe-133 and Xe 135 of ~0.2 mSv, and taking into account the contribution of HTO and I-131, it was obtained a the total committed effective dose of ~0.3 mSv. Finally, although the dose estimated was much lower than the limits and operating restrictions for workers, the causes that gave rise to the event are being analyzed.

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The authors wish to thank Ph.D. James Marsh for the paper provided and his valuable comments.

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OS2.3 (T2.2-0414)

MIODOSE: New Software to Integrate Uncertainty in the Optimisation of Monitoring for Internal Contamination

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In case of risk of occupational intakes of radionuclides, the potential internal contamination of workers must be monitored. This monitoring is carried out by measuring the activity retained in the body or present in excreta. The results of these measurements can be interpreted in terms of committed effective dose using biokinetic and dosimetric models adapted to exposure circumstances.

However, the measurement variability and the incomplete knowledge of exposure conditions introduce uncertainty in the dose assessment. Statistical methods were developed to evaluate this uncertainty as a criterion to optimize individual monitoring programs. The objective is to guarantee compliance with dose limits or dose constraints within a defined level of confidence and using reasonable operational means. These statistical methods were implemented in MIODOSE software developed in collaboration between IRSN and ORANO.

This software allows, by integrating uncertainty:

- estimating the minimum dose detectable by a routine monitoring program from available information on physico-chemical forms of the handled material, on the level of activity at the workplace, and on the detection limits of the techniques available to measure incorporated radionuclides;
- assessing the committed effective dose following an intake incident, along with its associated uncertainty, from measured retained and/or excreted activities;
- helping the person in charge to choose the monitoring program best adapted to the potential or ascertain exposure conditions.

The developed methods will be explained and applied to real cases in order to demonstrate their practical interest.

Keywords: Internal dosimetry, monitoring, uncertainty)

OS2.3 (T2.5-0371)

Evaluation of Uranium Biokinetics and Associated Toxicity in Human Body from Protracted Ingestion of Groundwater

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Uranium is the heaviest actinide, naturally prevailing in almost all rocks, soil, surface and groundwater. The health outcomes of uranium are the synergistic effect of its chemical and radiological toxic mechanisms. Groundwater samples collected from wide-ranging sites of granite-rich Bhiwani district of Haryana were analyzed for natural uranium content using LED fluorimeter. In this region, U concentration varied from 11.55 to 423.33µg L^{-1} with an average of 119.79 µg L^{-1}, exceeding the prescribed limit of 30 µg L^{-1} by WHO, 2011 and 60 µg L^{-1} by Atomic Regulatory Board, India. The average cancer mortality risk for ^{234}U, ^{235}U, ^{238}U was 11.942×10^{-10}, 3.97×10^{-8}, 6.67×10^{-6} and average cancer morbidity risk was 4.64×10^{-10}, 6.27×10^{-8}, 10.57×10^{-6} respectively. The mean values of Lifetime Average Daily Dose (LADD) and Hazard Quotient (HQ) were 2.36µg $kg^{-1}d^{-1}$ and 1.96 (<1) respectively, posing a considerable health hazard to the general public. Highest doses are imparted to infants of 7-12months, making them the most vulnerable group of population to U doses.

Hair compartment model proposed by Li et al., 2009 was employed for prospective and retrospective assessment of U retention and the calculation of tissue and organ doses following chronic intake for 60 years via drinking water pathway. Hair growth rate was normalized at 0.1g d^{-1} and alimentary tract factor as 0.6%. Cortical bone volume and soft tissue form the stable U repository in the human body. Bone surface (38%), kidneys (14%), LLI (11%), ULI (5%) and liver (5%) are the major dose recipients. The annual effective dose to the whole body due to U ingestion via drinking water ranges from 1.92µSv to 70.32µSv with an average of 19.90µSv which is below the threshold of 100µSv (WHO, 2011). The mean uranium excreted from the body per day via faeces, urine and hair were 166.71µg, 0.35µg and 0.64µg, respectively.

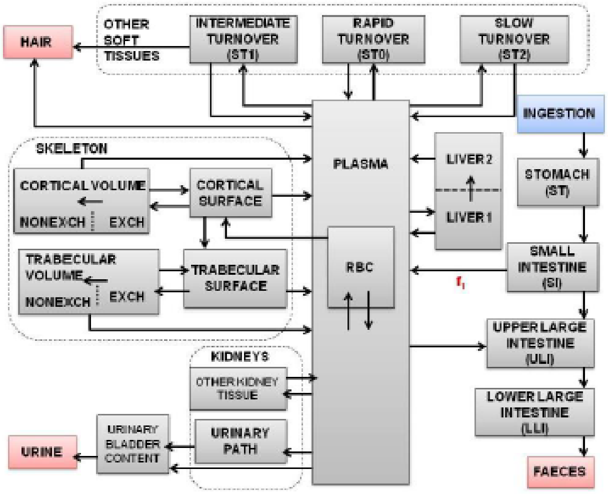

Fig. 1. Hair compartment model of uranium

Keywords: Uranium biokinetics, Hair compartment model, Age and organ-specific doses

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OS2.4 (T2.4-0673)

Biological Dosimetry - Status and Challenges

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In a radiological or nuclear emergency it is highly important to obtain personal information about a potentially received radiation dose and thus to take fears or, if necessary, initiate individually tailored treatments. However, retrospective dose estimation, especially in case of large scale events with a large number of possibly exposed individuals remains a challenge. This does not only apply to retrospective dose estimation but also to accompanying measures such as collection and transport of biological samples or personal devices. During the years several strategies have been developed, tested and are trained to be prepared for different scenarios. Today there is a broad and increasing spectrum of assays, which can be used for retrospective dose estimation and depending on the circumstances the best strategy can be chosen to the benefit of the individual person, but also for society. Thus, even after weeks, a supposedly high radiation exposure can be individually excluded and a "worried healthy" person with supposed radiation symptoms can be differentiated from an actually exposed person.

Technical development for the analysis of approved, validated biomarkers has been driven forward since years with great success. Today, due to automation processes, the long established cytogenetic biomarkers are still very well suited for larger scale events. In addition new molecular biological biomarkers are on their way to be integrated in biological dosimetry, e.g. for a fast classification of individuals or to identify persons who will need medical care.

Networking between laboratories has also shown to have a high potential for handling mass causalities. This strategy is followed worldwide and there are specialized active networks in America, Asia, and Europe. This strategy is also supported by global organisation, as WHO and IAEA. Besides preparedness for different emergency scenarios, biological dosimetry and the applied methods also contribute to radiation research topics, e.g. on low dose research or individual radiation sensitivity.

An overview will be given about established and upcoming assays as well as regional and global networks and their activities. Presented will be also initiatives, which actively or indirectly support the activities and preparedness of these networks, either by bringing together relevant organisations or by supporting the transfer of knowledge and skills.

Keywords: biological dosimetry, networking, large scale analysis

ACKNOWLEDGMENTS

We thank all experts in this field, especially all partners of the Latin American network, American Network, the Asian Networks, European Network, the WHO BioDoseNet and the experts in the high throughput field. All together form a highly ambitious community with the goal of being prepared for an event that everyone hopes will never befall anyone.

OS2.4 (T2.1-0302)

Biological Dosimetry Assays Applied to Radiosensibility Studies in Pediatric Patients with Primary Immunodeficiency

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Individual radiosensitivity (RS) is a characteristic inherent to the subject associated with an increased reaction to ionizing radiation (IR), influenced by genetic susceptibility. Primary immunodeficiencies (PIDs) are genetic diseases with defects in the development, function and / or regulation of one or more components of the immune system. A group of PID is characterized by defects in DNA repair systems with increased RS. Numerous studies have been carried out in order to assess whether there is a correlation between clinical radiosensitivity and cellular radiosensitivity determined by in vitro tests on fibroblasts and lymphocytes.

The cytokinesis block-micronucleus test (CBMN) and the comet assay, single-cell gel electrophoresis (SCGE), would allow the evaluation of this RS. The potential utility of RS predictive tests is particularly linked to planned medical exposures, as an additional tool for patient radiological protection and the administration of personalized treatment.

Three clinical cases of pediatric patients with PID are described: Case 1) 1 year old girl with inflammatory bowel disease, combined immunodeficiency without pathogenic variants in molecular biology panel for PID, evaluated in 2018 by CBMN. Case 2) a 9-year-old girl, with antecedents of recurrent respiratory infections and HyperIgM syndrome, which may associate alterations in DNA repair, evaluated in 2012 by SCGE. In 2018 diagnosed with activated PI3Kδ syndrome (pathogenic gain-of-function variants in the genes encoding phosphoinositide 3-kinase δ), a RS study was repeated using CBMN. Case 3) 15 years old boy with two heterozygous mutations of gen PRKDC associated with immunodeficiency, granuloma, uveitis, lymphocytopenia and leukopenia, evaluated in 2019 by CBMN.

Results: Case 1) The frequency of micronuclei (MN) shows values compatible with normal RS, Case 2) the RS studies conducted in 2012 and 2018 showed hypersensitivity to IR. Case 3) the MN frequency values could be related to the heterozygous mutations observed in the PRKDC gene and the hypersensitivity associated with an increased reaction to IR.

Conclusion: These results show the importance that predictive trials ‘application would provide for the evaluation of patients with suspected / diagnosed PID who may require radiant procedures or radiomimetic drugs for complications (autoimmunity, inflammation, malignancy) or hematopoietic precursor transplantation.

Keywords: Radiosensitivity, immunodeficiencies, DNA repair.

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OS2.4 (T2.1-0318)

Recent Developments on Automated High Throughput Biodosimetry Tools for Radiological/Nuclear Mass Casualty Incidents

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In the case of radiological /nuclear (R/N) mass casualty incidents, several hundreds and thousands of humans may be exposed to ionizing radiation. Timely assessment of radiation dose is critical for making an appropriate "life-saving" clinical decision for those exposed to moderate and high doses of radiation. Currently available biodosimetry techniques are time consuming and laborious making them impractical for mass casualty incidents. Therefore, it is imperative to develop automated high throughput platforms for some of the well-known biodosimetry tools to provide rapid biodosimetry. An equally important aspect is to discover new biomarkers that can enhance the rapidity of dose assessment on samples obtained with minimal invasion. To achieve this objective, a multidirectional approach has been undertaken at the REAC/TS Cytogenetic Biodosimetry Laboratory: (I) Development and automation of high throughput Dicentric Chromosome and CBMN assays on robotic platforms, (II) Increase the rapidity of dose assessment by automated image analysis (III) Automated analysis of a novel biomarker, Pseudo-Pelger Huet Anomaly (pseudo PHA), (IV) Development of automation for the Premature Chromosome Condensation (PCC) technique and (V) Development of electronic training tools to teach and train clinical laboratorians for increasing the surge capacity of dicentric chromosome scorers for validation of estimated radiation dose. Additionally, efforts are being made to optimize the interphase chromosome breakage analysis (ICBA) tool that can be efficiently performed on any human cell type for dose assessment. Optimization and clinical validation of high throughput techniques in robotic platforms will enable a rapid assessment of absorbed radiation dose for several hundreds and thousands of exposed people. Our multidirectional approach using automated high throughput robotic platforms will constitute an efficient radiation emergency response to fulfill the biodosimetry needs of R/N casualty incidents in the future.

Acknowledgements: The funding received from the US Department of Energy/National Nuclear Security Administration is gratefully acknowledged.

OS2.4 (T2.1-0453)

Clinical Validation of a High Throughput, Direct from Blood, Gene Expression based Biodosimetry Test

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The CE-IVD marked REDI-Dx test is a direct from stabilized blood, gene expression based high throughput biodosimetry test system undergoing clinical validation in preparation for a de novo submission to the US FDA. The test system consists of a 1.0 ml draw blood collection tube, a multiplex gene expression assay kit, and the Thermo Fisher ABI 3500xL Dx Genetic Analyzer with custom analysis software. The blood samples are stable for 14-days at ambient conditions for transport and the testing process takes as little as 6 hours with a throughput of up to 1100 samples per 24 hour day per system. The test is designed to estimate dose from 0 to 10 Gy. Test accuracy as well as sensitivity and specificity are being evaluated using a combination of blood samples from human cancer patients undergoing total body irradiation (TBI) in preparation for bone marrow transplant, TBI non-human primate (NHP) samples, and more than 500 non-irradiated human samples including burn, trauma, influenza, and other potentially confounding conditions. There were zero false positives in the human variability samples consisting of Caucasian, African American, Hispanic and Asian adults, as well as geriatric and pediatric samples with no apparent effect of age, race or sex. Similarly, in the potential confounder cohort representative of common chronic conditions, pregnancy, lactation, obesity, and likely therapeutics, there were zero false positives. Granulocyte-colony stimulating factor (G-CSF) does lead to false positives after 5 days of daily treatment, so sample collection should be performed prior to or at the time of administration. Testing of 44 influenza patients yielded 1 patient with an estimated dose of above 0.5 Gy, and burn, severe infection and trauma evaluations are underway. Several pre-irradiation cancer patients were reported as false positives with a mean estimated dose of 0.2 Gy. Initial results from Human TBI samples (141 fractionated dose samples, 3 x 1.2 Gy, Nominal doses of 3.6, 7.2, and 10.8 Gy) showed good separation between dose levels and a correlation of r= 0.93 with nominal dose. Human TBI results will be compared to matched regimen fractionated dose NHP TBI to bridge between humans and NHPs. NHP based single dose TBI accuracy assessment is on-going. The REDI-Dx Biodosimetry Test System is For Investigational Use Only in the USA and bears CE-IVD marking in EFTA and EU.

Keywords: Biodosimetry, high throughput

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OS2.4 (T6.4-0313)

Advances in Biological Dosimetry

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Biomarkers are essential in evaluating exposure and for predicting future adverse health outcomes. Development of cellular and molecular biomarkers is an important goal in public health research. Many attempts have been made to identify biomarkers which can be used for high throughput biodosimetry screening. Chromosome alterations such as dicentrics and translocations induced by ionising radiation can be detected by using fluorescence in situ hybridisation (FISH) using telomere and centromere specific peptide nucleic acid (PNA) probes as well as by multi-colour FISH. DNA damage can be measured by both γH2AX and micronuclei analysis. Gene expression profiles using microarray analysis will be useful in identifying signature genes of exposure to ionising radiation. Predictive genomics may be a promising approach to high-throughput radiation biodosimetry. It is anticipated that such a multiparametric approach using advanced biodosimetry techniques may be needed in estimating the biologically relevant doses of radiation in accidental or occupational exposures. The application of different assays – classical cytogenetic assays and molecular biodosimetry assays – will be discussed in the presentation with reference to low- and high-LET radiation exposure

Keywords: Ionising radiation, high and low-LET, Biological Dosimetry

OS2.5 (T2.3-0374)

Contact Lens-Type Ocular In Vivo Dosimeter for Radiotherapy

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This study aimed to (i) develop a contact lens-type ocular in vivo dosimeter (*CLOD*) that can be worn directly on the eye and (ii) assess its dosimetric characteristics and biological stability for radiation therapy. The molder of soft contact lens was directly used to make the dosimeter, which included a radiation-sensitive component—an active layer similar to a radiochromic film—to measure the delivered dose. A flatbed scanner with reflective mode was used to measure the change in optical density due to irradiation. The sensitivity, energy, dose rate, and angular dependence were tested in this study and the uncertainty in determining the dose was calculated using an error propagation analysis. Sequential biological stability tests, specifically, cytotoxicity and ocular irritation tests, were conducted to ensure the safe application of *CLOD* to patients. The dosimeter demonstrated high sensitivity in the low dose region, which linearly decreased with the dose. The differences for the 10 and 15 MV photon beams were 1.7% and 1.9%, as compared to the 6 MV photon beam, which increased with energy. No significant dose rate dependence was obtained for the *CLOD*. Angular dependence was observed from 90° to 180° with difference of response from 1% to 2%. The total uncertainty using error propagation analysis decreased as a function of dose with the red channel, presenting uncertainty values of 11.0%, 5.6%, 2.4%, 1.4%, and 0.9% at 5, 10, 20, 30, and 50 cGy, respectively. Quantitative evaluation using the MTT method presented no cytotoxicity. Further, no corneal opacity, iris reaction, or conjunctival inflammation were observed. The *CLOD* is the first dosimeter that can be worn close to the eye. The results of cytotoxicity and irritation tests indicate that it is a stable medical device. The evaluation of dose characteristics in open field condition shows that the *CLOD* can be applied to an in vivo dosimeter in radiotherapy.

(a) Sandwiched structure and design of *CLOD* (b) Manufactured *CLOD*

Fig. 1. (a) Sandwiched structure and design of *CLOD* and (b) manufactured *CLOD*

Keywords: *Contact lens dosimeter, Lens dose, Radiation therapy*

ACKNOWLEDGMENTS
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OS2.5 (T2.2-0415)

Automatic Segmentation of Cardiac Structures and Dosimetry Evaluation for Breast Cancer Radiotherapy

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Cardiac toxicity from adventitious irradiation of the cardiac substructures during radiation therapy can lead to a variety of cardiac complications. Several studies have discussed the radiation dose-volume effect to cardiovascular diseases in patients (1). However, the mechanisms of radiation-induced cardiac toxicity and cardiac structures involved are not clearly understood and data showing the relationship between radiation dose to heart substructures and subsequent cardiac complications are scarce.

We developed an automatic method to segment cardiac sub-structures given in a radiotherapy planning computed tomography (CT) image set to support epidemiological studies or clinical trials looking at cardiac disease endpoints after radiotherapy. Our method is based on a library of 30 detailed cardiac atlases combined with a most-similar atlas selection algorithm and three-dimensional (3D) deformation. We cross-validated our method within the library by evaluating standard geometric comparison metrics (i.e., Dice Similarity Coefficients (DSC) and Average Surface Distance (ASD)) and by comparing cardiac doses for simulated breast radiotherapy between manual and automatic contours. We also analyzed the impact of the number of cardiac atlas in the library and the use of manual guide points on the performance of our method. The DSCs from the cross-validation reached up to 97% (whole heart) and 70% (chambers). The ASD for the coronary arteries was less than 11.0 mm, with the best agreement (7.3 mm) in the Left Artery Descending (LAD). The dose comparison for simulated breast radiotherapy treatments showed absolute dose differences less than 0.3 Gy for the whole heart, whereas the coronary arteries showed differences less than 2.3 Gy (LAD). The sensitivity analysis showed no notable improvement in our method beyond ten atlases and the use of manual guide points does not significantly improve performance.

The developed method for automatic segmentation of cardiac substructures for radiotherapy CTs, combined with accurate dosimetry techniques, should be useful for cardiac dose reconstruction of a large number of patients in epidemiological studies or clinical trials.

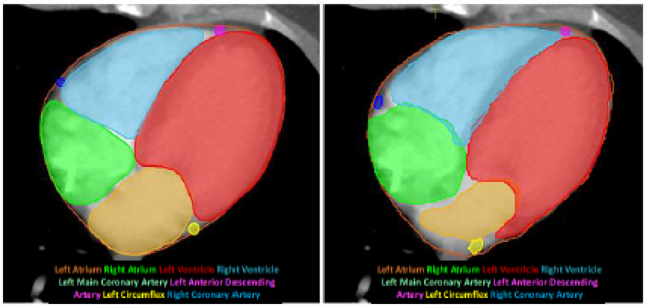

Figure 1. Cross sectional view of manually-drawn (left) and automatically-generated (right) heart substructures

Keywords: Cardiac structures, Automatic segmentation, Deformation, Breast radiotherapy

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OS2.5 (T2.2-0684)

Experimental validation of TG-43 brachytherapy dose calculation formalism for HDR Co-60 brachytherapy source

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This study has been designed to develop a three-step method of in-phantom dosimetry around a HDR brachytherapy Co-60 photon source. The present development of a method of in-phantom dose measurement under reference HDR brachytherapy conditions was performed in recognition of the primary role of brachytherapy dose calculations according to the AAPM TG-43 recommendations. This also can be considered as an experimental validation of TG-43 consensus data. This method is a frame for in-phantom dosimetry in brachytherapy, analogous to "small calibration field" commonly used in teletherapy to provide the reference conditions for the cross-calibration of high-resolution detectors. The first step was to measure the absorbed dose rate to water with the calibrated ionization chambers and with solid-state detectors such as silicon diodes and diamond detector under reference conditions. The second was to determine Radial and Anisotropy function. Under these identical reference conditions, third step was to cross-calibrate, small solid-state detectors such as silicon diodes, synthetic diamond detector well suited for spatially resolved dose rate measurements particularly at smaller source axis distances in the water phantom. In water dose rate measurements were performed using PTW semiflex chamber type 31010, PTW semiflex chamber type 31013, PTW Pinpoint chamber type 31006, PTW microdiamond type 30019 and PTW Diode detector 60012 (PTW Frieburg, Germany). Eckert & Ziegler BEBIG's "SagiNova" model of HDR Brachytherapy Co-60 source machine with PTW MP3 Scanning water phantom and "MEPHYSITO" software were employed for the measurements. The distance of 40 mm from the source axis was chosen as the reference condition for cross calibrations. The correction factors have been devised, developed and evaluated during the experimental measurements with various dosimeters. The results of present study showed that major contributing correction factors were volume averaging and position of the chamber's effective point of measurement (EPOM) during the validation. The present work provided methodology for (1) the experimental determination of the position of the source axis, (2) a general formulation for the volume averaging correction factor of small ionization chambers and (3) the experimental determination of the EPOM positions. Detailed results including correction factors specific to particular kind of detector and cross calibration values will be discussed further during meeting.

Keywords: Brachytherapy, Cobalt-60, phantom measurement

ACKNOWLEDGMENTS

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OS2.5 (T2.1-0244)

Assessment of radiation dose to eye lens during Rapid Arc treatment of Head and Neck cancer patients

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Aim: The present study was aimed to assess radiation dose to eye lens for the head and neck cancer patients treated with Rapid Arc^{TM}.

Material and Methods: Twenty patients with head and neck cancer were randomly assigned to participate in the present study after obtaining a written informed consent. The inclusion criteria of patient in study were having disease extended bilateral and a curative intent. The patient age (range, 38 - 65 years; mean, 51 years) treated with Volumetric Modulated Arc Therapy (VMAT)/ Rapid Arc^{TM} which is a form of external beam radiation therapy (EBRT). All the patients were planned with Eclipse treatment planning system version 13.7 (Varian Medical Systems, Inc., Palo Alto, USA) and treated with RapidArc^{TM} dual arc (1 isocenter, 2 full arc, ±30° collimator angle) technique using Trilogy (Varian Medical Systems, Inc., Palo Alto, USA) linear accelerator (Linac) equipped with 60 pair Millennium Multi-Leaf Collimator (MLC). The patients treated with a conventional fractionation regime of 70Gy/35 fractions, with a dose delivery of 2 Gy/fraction. The eye lens doses were assessed by placing the dosimeter as close as possible to the eye in contact with orfit. The OSL dosimeters were from Landauer Inc., Al_2O_3:C nanoDots^{TM} (10 X 10 X 2 mm). A set of information were recorded from each of the participants such as patient ID, patient characteristics, age, skull size, distance of planning target volume (PTV) from eyes, target volume, Monitor units (MU) delivered during treatment etc.

Results: The total target volume was recorded in range, 338 – 560 cc; mean, 460 cc. The distances of planning target volume (PTV) edge from eyes were found in range, 3.0 – 8.7 cm; mean±SD, 4.5±2.8 cm. It was observed that maximum eye lens dose was received during the treatment of cancer of the maxilla. This was measured 1.26 cGy per fraction for a mean dose delivery of 200 cGy/ #, i.e. 0.63% of the tumor dose. At the end of the EBRT, the cornea would have received total estimated dose of 44.10 cGy in 7 weeks. Whereas during the treatment of Ca Vocal cord was responsible for the minimum corneal mean dose 1.14 cGy per fraction for a dose delivery i.e. 0.57% of the tumor dose. At the end of the EBRT, the eye lens would have received total estimated dose of 39.90 cGy in whole treatment fractions. It was observed that minimum eye lens dose was received due to greater distance of eye from PTV edge. There was no significant correlation observed in eye lens doses with respect to target volume and average MUs delivered during Rapid Arc^{TM} treatment. The average cumulative dose to eye lens was estimated 42 cGy with Linac Rapid Arc^{TM} treatment of head and neck cancers.

Conclusion: The radiation dose to eye lens is critical and important. Our results show that assessment of dose to eye lens is recommendatory during radiotherapy treatment of curative cancer patients.

Keywords: Eye lens dose, Rapid Arc, Optically stimulated luminescence (OSL) dosimetry

OS2.6 (T2.1-0531)

OpenRadiation: a collaborative project for radioactivity measurements in the environment by the public

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After the Fukushima accident, initiatives emerged from the public to carry out themselves measurements of the radioactivity in the environment with various devices, among which smartphones, and to share data and experiences through collaborative tools and social networks. Such measurements have two major interests, on the one hand, to enable each individual of the public to assess his own risk regarding the radioactivity and, on the other hand, to provide "real time" data from the field at various locations, especially in the early phase of an emergency situation, which could be very useful for the emergency management.

The objective of the OpenRadiation project is to offer the public the opportunity to perform measurements of the radioactivity using connected dosimeters on smartphones. The challenge is to operate such a system on a sustainable basis in normal situations and being useful in an emergency situation. In normal situations, this project is based on a collaborative approach with the aim to get complementary data to the existing ones, to consolidate the radiation background, to generate alerts in case of problem and to provide education & training and enhanced pedagogical approaches for a clear understanding of measures for the public. In case of emergency situation, data will be available "spontaneously" from the field in "real time" providing an opportunity for the emergency management and the communication with the public.

The practical objectives are to develop i) a website centralising measurements using various dosimeters, providing dose maps with raw and filtered data and offering dedicated areas for specific projects and exchanges around data and ii) a dosimetric app using a connected dosimeter. This project is conducted within a partnership between organisms' representative of the scientific community and associations to create links with the public.

This project is conducted within a partnership between organisms' representative of the scientific community and associations to create links with the public. The website is available since October 2017 and more than 100 000 measurements were performed by about 80 users.

OS2.6 (T2.1-0281)

Current status of a carborne gamma-ray survey system, KURAMA-II

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KURAMA-II, a carborne gamma-ray survey system characterized as its compactness, autonomous operation, acquisition of pulse height spectrum data along with GPS data, has established its position as an effective method for the radiation monitoring method in the environment on a long-term basis[1]. Periodical surveys in eastern Japan have been conducted by the Japanese government since 2011. A continuous carborne survey of radiation by KURAMA-II installed on local buses continues in Fukushima prefecture as a collaboration among Fukushima prefecture, Kyoto University, and JAEA.
The applications of KURAMA-II have begun to extend beyond just monitoring the radiation. One of such examples is the implementation of KURAMA-II into a robot for the recovery of farmlands near Fukushima Daiichi Power Plant. This robot is also equipped with a hyperspectral system for fertility estimation, a high-precision GPS-based guidance system, and the data will be processed at a cloud-based analysis and visualization system. Now the implementation of KURAMA-II is finished and a series of field test including the comparison with the results of a standard procedure in soil-contamination measurements is on the way. A certain number of this kind of robots will be deployed intending to accelerate the recovery and farmlands, especially in difficult-to-return zones around Fukushima-Daiichi Power Plant.
Another trial is to port the system to a low-power single-board computer for realizing a low-power, compact KURAMA-II as a tool for the prompt establishment of radiation monitoring in a nuclear accident. Autonomous network communication based on LPWA (Low Power Wide Area network) is introduced to this type of KURAMA-II, realizing the extensive coverage of the radiation monitoring network under extreme situations such as a large-scale black-out caused by an earthquake.
The present status and prospects of KURAMA-II, including the status of these trials, are introduced.

***Keywords**: KURAMA-II, soil contamination, single-board computer*

ACKNOWLEDGMENTS
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OS2.6 (T2.5-0430)

Key lessons learned from the detection of ^{106}Ru in Hungary in the fall of 2017

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Anthropogenic radioruthenium (^{106}Ru) has been detected in the environment from late September to early October 2017 by the environmental monitoring networks of several European countries, including the monitoring stations of the Hungarian National Environmental Radiological Monitoring System (NERMS) [1]. The widespread, continental scale detection made the comprehensive analysis of the available monitoring data necessary. However, difficulties arose in the direct comparison of the reported data due to the different length of sampling periods in relation to the residence time of the contaminant. This required the mathematical unification of the variant sampling periods for their usage as integration periods [2]. The evaluation of the measurement data was further complicated by the availability of data with deficient information content and different data format.
This paper's objective is to assess the data supply procedure of the Hungarian environmental radiological monitoring systems based on the key lessons learned from the ^{106}Ru detection event. General proposals were determined for the evaluation of individual measurement data applied e.g. in the uncertainty analysis and improvement of detection limits of environmental measurements, as well as in the reporting of measurement results. Furthermore proposals were developed for the statistical analysis of environmental radiation monitoring, including the comparison of data originated even from different entities.
As a result of this work, the accuracy of the monitoring data can be increased and their usage in the assessment of dose consequences may be improved in a similar release case.

Keywords: ^{106}Ru release, environmental monitoring, statistical analysis

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OS2.6 (T2.1-0043)

Natural Radiation exposure to the public in Douala city, Cameroon

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In order to study natural radiation exposure in Douala, the largest and most populated city in Cameroon, in-situ radiation measurements were carried out with NaI(Tl) scintillation detector at 1m in height from the soil ground to determine activity concentration of ^{238}U, ^{232}Th and ^{40}K. Also, the car-borne survey was conducted in Douala to measure air absorbed dose rates. Indoor radon and thoron concentrations were measured in 71 dwellings in Douala city using passive type radon-thoron detectors (RADUET) for long-term measurements. Thoron progeny concentrations were also measured with deposition detectors placed with the radon-thoron detectors. Moreover, soil samples were collected in a part of Douala for radioactivity measurements in the laboratory. From the results of the measurements, arithmetic mean, minimum and maximum air absorbed dose rates estimated to be 50 nGy h^{-1}, 29 and 86 nGy h^{-1} respectively. This arithmetic mean is lower than the world average value of 59 nGy h^{-1}. The activity concentrations obtained using NaI(Tl) detector varied from 18 to 47 Bq kg^{-1} for ^{238}U, 21 to 54 Bq kg^{-1} for ^{232}Th, and 10 to 410 Bq kg^{-1} for ^{40}K with averages of 29, 38, and 202 Bq kg^{-1} respectively, for in-situ measurements. They vary between 29-98 Bq kg^{-1} for ^{238}U, 29-92 Bq kg^{-1} for ^{232}Th, and 40 to 79 Bq kg^{-1} for ^{40}K, with the averages of 60, 57, and 56 Bq kg^{-1} respectively for soil samples collected at Douala III subdivision. However, radon, thoron and EETC were respectively found to vary from 31 ± 1 to 436 ± 12 Bq m^{-3}, 4 ± 7 to 246 ± 5 Bq m^{-3}, and 1.5 ± 0.9 to 13.1 ± 9.4 Bq m^{-3}. The arithmetic mean values of radon, thoron and thoron progeny concentrations were respectively found to be 139 ± 47 Bq m^{-3}, 80 ± 52 Bq m^{-3}, and 4.6 ± 2.9 Bq m^{-3}. The mean value of the equilibrium factor for thoron is estimated at 0.11 ± 0.16. The highest annual external effective dose of 0.7 mSv y^{-1} for in-situ measurements by car borne survey observed at Ndogbong is higher than the world average value of 0.5 mSv y^{-1}. The annual effective dose due to exposure to indoor radon and progeny was found to vary from 0.6 to 9 mSv y^{-1} with an average value of 2.6 ± 0.1 mSv y^{-1} and the effective dose due to the exposure to thoron and progeny was found to vary from 0.3 to 2.9 mSv y^{-1} with an average value of 1.0 ± 0.4 mSv y^{-1}. The contribution of thoron and its progeny to the total inhalation dose was found to vary from 7 to 60% with an average value of 26%; thus their contributions should not be neglected in inhalation dose assessment.

In general, radiation doses have shown no significant health risk due to exposure to natural radiation from external sources, to radon, thoron and their progeny in the study area.

Keywords: Natural radioactivity, air absorbed dose rate, radon/thoron, progeny.

ACKNOWLEDGMENTS

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OS2.6 (T2.3-0258)

The EPRI Demonstration of an Autonomous Site Characterization Vehicle

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Performing radiological characterization of structures and land areas during decommissioning of a nuclear facility involved multiple surveys of land areas and large structures to support remedial action and then and eventual Final Status Survey for site release. These surveys are typically performed using manual methods, which is time consuming and labor intensive. Due to the repetitive effort of performing surveys over large areas, automation of radiological characterization techniques has a high potential to be successful and promises substantial benefit.

EPRI has performed a project that started with the development and then the eventual demonstration of an autonomous site characterization system. This EPRI project focused on building surfaces and the surfaces of large land areas as these configurations lend themselves more readily to automated survey. The project used an existing EPRI robot platform, but included the development of a LIDAR mapping program to define the limits of the robot pathways, and to provide location information about the robot when inside. A GPS provides the location when outside.

Figure 1 The EPRI robot with LIDAR unit in the front, GPS on the top. Radiation sensors include dual NaI spectrometers on the sides, and a doserate meter.

The radiation sensors were supplied by Mirion Technologies – Canberra, and included an LED-stabilized 3x3 NaI detector, Osprey MCA, ISOCS efficiency calibration, and a Data Analyst to repeatedly collect and analyze each 3-second spectrum. Nuclide results are obtained each 3 second. Dual sensors were deployed spaced 1 or 2 meters apart to reduce the survey time. The EPRI geo-mapping algorithms are then used to generate nuclide-specific contour maps.

The system was successfully demonstrated at the Kewaunee nuclear power plant site in the fall of 2019. The demonstration included autonomous surveys of floor areas inside the Auxiliary Building, and autonomous surveys of environmental areas where potentially radioactive items were stored in the past.

Keywords: Robot, Mapping, Spectrometry

OS2.6 (T2.3-0323)

In-Situ Sample Analysis with Portable Gamma Spectrometers

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The conventional method for monitoring specific isotopes of interest involves taking swipes and bringing those samples to a laboratory environment for counting. This offers two advantages: stationary laboratory detectors are often more sensitive than devices used in situ and the ambient radiation levels are expected to be lower in a counting laboratory than in a potential radiological area where samples are taken. There are already cases when laboratory sample analysis is not necessary, for example in an area that is not expected to have any radiation a survey can be done with a portable scintillator to identify unexpected isotopes. However, the limits on spectral resolution of portable equipment have complicated efforts to search for the presence of isotopic signatures that are weak relative other isotopes present. This work will focus on the application of recently developed portable gamma spectrometers to in-situ radiological measurements.

There are many real world environments where radiation levels are relatively high but the presence of a particular isotope or class of isotopes is problematic, even in small quantities. One example would be the presence of fission products in the primary coolant of a nuclear power plant. It may be challenging to observe the peak from Cs-137 in the presence of strong Co-60 sources, which provide a continuum that Cs-137 must be detected on top of, or an Ag-110m source which has a strong peak 5 keV below the Cs-137 photopeak. A high resolution spectrometer, such as 3D position sensitive CdZnTe (3DCZT) devices commercially available with total crystal volumes on the order of 19cm^3 and energy resolution of 1% FWHM at 662 keV, will be capable of achieving a lower detection threshold for Cs-137 in this environment than a handheld scintillator with energy resolution on the order of 3%-7% FWHM. This work will study the use of portable high resolution gamma spectrometers for in situ counting applications and compare their sensitivity to conventional portable spectrometers.

There are various configuration options that are unique to 3DCZT. For example, an application specific integrated circuit (ASIC) is required for event readout and the choice of ASIC plays a major role on device performance. There are currently two different ASIC configurations available commercially from H3D, one has better energy resolution and the other has lower dead time. These two ASICs will be compared for similar applications to determine when it is ideal to use one technology over the other. Another major capability of 3DCZT is to discriminate based on event type. For example, events in the central pixels of the detector have a higher peak-to-Compton ratio than pixels near the edge of the detector. Also, events that trigger only one pixel tend to have better energy resolution while events that trigger more events tend to have better peak-to-Compton ratio. The ideal configuration to maximize sensitivity will be discussed based on the in-situ measurement scenario.

Keywords: *Gamma-ray spectroscopy, sample analysis, CdZnTe, CZT*

OS2.6 (T2.5-0345)

Radioactivity-induced Charging: Theory and Measurements

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Nuclear events Chernobyl disaster in 1986 and Fukushima Daiichi accident in 2011 can release large amounts of radionuclides into the environment and consequently cause substantial radiation damage to humans and ecosystems. The scope and the level of the radiation damage depend on the amount of radioactivity accumulated in the environment. Accumulation of radioactivity mainly results from atmospheric dispersion and deposition of radioactive particles, indicating that predicting atmospheric behavior of radioactive particles is a key step to assess the impact of such nuclear events on public health and environmental pollution.[1]

Attempts have been made to investigate the behavior of individual radioactive particles and radioactive particle populations in air.[2-4] These studies have shown that the behavior of radioactive particles can significantly change over time because of strong electrostatic particle-particle interactions. When radionuclides decay into different nuclides, radioactive particles can instantly acquire and accumulate electrical charge by self-charging and diffusion-charging mechanisms. Self-charging is due to acquisition of electrical charge via the emission of alpha and beta particles. Diffusion charging is due to acquisition of electrical charge by capturing gaseous ions. Charge accumulated on radioactive particles can create strong electrostatic forces between the particles, affecting their microphysical processes. Thus, it is essential to better understand the charging mechanisms of radioactive particles in order to more accurately predict their atmospheric behavior and transport.

This research focuses on reviewing the progress of theoretical and experimental investigations of radioactivity-induced charging, including the authors' work. Theoretical approaches reviewed in this study include charge and force balances that have been studied since mid-1950s. Charge balance equations consist of terms describing charging mechanisms, such as self and diffusion charging. Force balance equations generally include electrodynamic forces to consider radioactivity-induced charging. Various measurement techniques to quantify radioactivity-induced charging are introduced. Advantages and disadvantages of measurement techniques of radioactivity-induced charging are discussed. Application studies of radioactivity-induced charging are also examined.

Keywords: *Radioactive particle charging, Radioactivity transport, Nuclear power plant accidents*

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OS2.6 (T2.5-0457)

Influence of Radioactive Decay on Atmospheric Transport of Radioactive Particles

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Human health and environmental risk assessments of radiation exposure, after a nuclear plant accident or a deliberate explosion of a radiological dispersion device, are based on theoretical predictions of atmospheric dispersion of radionuclides. However, because of large discrepancies found between theory and observations after nuclear accidents, improving predictive modeling of radioactivity transport is a top priority.[1] In general, it is assumed that the microphysical processes of atmospheric particles containing radionuclides are independent of electrostatic surface interactions.[2-3] Radioactive decay of radionuclides, however, may induce significant surface charging; therefore, electrostatic particle-surface interactions should be included in predictive models of radioactivity transport to reduce uncertainty in radiation risk assessments.

This study is focused on investigating the influence of radioactivity-induced charging on atmospheric dispersion of radioactive particles. A theoretical framework has been developed to incorporate particle charging and subsequent charge effects, chemical transformation of radionuclides, and microphysical processes of radioactive particles,[4-5] which are relevant to this investigation. It has been found that radioactivity-induced charging can significantly influence various microphysical processes of radioactive particles because the particles can acquire electrical charge immediately after radioactive decay of radionuclides. The acquisition of charge by the particles can create strong electrostatic particle-particle interactions. For example, the coagulation of highly radioactive particles can significantly be suppressed because strong electrostatic repulsive forces can be generated between the particles which can instantly acquire many positive charges by radioactive decay of numerous radionuclides. Then, the framework was incorporated into a transport model to assess the effects of radioactivity-induced charging on atmospheric dispersion of radioactive particles. Simulation results have shown that growth rates of radioactive particles dispersed in air can significantly change due to effects of radioactivity-induced charging, and this change can impact the deposition rates of the particles. Results of this investigation suggest that radioactivity-induced charging should be taken into account to more accurately predict transport of radioactivity after a nuclear event.

***Keywords**: Radioactive particle charging, Radioactivity transport, Nuclear power plant accidents*

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OS2.6 (T2.2-0647)

Estimation of Doses Possibly Received by Aircrews from a TGF using Two Different Theories of Production

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In 1994, a new type of gamma radiation coming from the Earth's atmosphere has been discovered. This radiation, now referred to as terrestrial gamma-ray flashes (TGFs), comes in the form of bursts of high-energy photons, lasting less than one millisecond [1]. Initially expected to be rare events, we now know that they are related to common thunderstorms [2 ; 3] and common types of discharges, namely intra-cloud lightning [e.g., 4]. Briggs et al. [5] estimate that 400,000 TGFs per year occur globally, as detectable by the Fermi Gamma ray Burst Monitor (GBM). Thunderstorms also produce another type of events, called gamma ray glows. These are significant elevations of the background radiation, lasting from seconds to minutes. Both are explained by the bremsstrahlung emission from relativistic runaway electron avalanches (RREA) [e.g., 6 ; 7] taking place in large potential differences in thunderclouds. A conservative estimate for the occurrence of gamma-ray glows is that 8% of electrified storms produce them [6].

Both TGFs [e.g., 8] and gamma ray glows [e.g., 9] are produced at thunderstorm altitudes, which match with flight altitudes of commercial flights. That calls for a precise assessment of the risk encountered by aircrew and passengers on a plane that would fly near or through thunderstorms. The exposure of aircrew to cosmic radiation is monitored by software that estimates the dose received for each individual, within the route flight data, taking into account the galactic component (cosmic rays) and relevant solar flares. Maximum annual doses can reach 6 mSv. In 2010, Dwyer et al. [10] already estimated the dose that a TGF could produce as a function of the diameter of the electron beam, and showed that TGFs could be an additional non negligible exposure for aircraft passengers. Since then, and as these events are not well understood yet, no other studies have been carried out to estimate the risk for aircraft passengers. If TGF doses are assessed as significant, this could challenge the paradigm of dose calculation assessment and require the implementation of on-board measurement systems, which would have a significant cost for airlines.

In this work, as a first step in the risk evaluation, we will present calculations of doses produced by TGFs and related processes: RREA, gamma-rays and secondary particles, that would be received by humans if an airplane were to find itself in such an event. Calculations for electron doses will be realized within two different theories of production of TGFs (leader-based model [e.g., 11 ; 12] and relativistic feedback model [e.g., 13]).

Keywords*: Terrestrial Gamma ray Flashes, Aircrew Dosimetry, Thunderstorms*

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OS2.7 (T3.4-0309)

Development of Pediatric Mesh-type Reference Computational Phantom Series of International Commission on Radiological Protection

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Since its 2007 Recommendations, the International Commission on Radiological Protection (ICRP) has used the voxel-type reference computational phantoms (VRCPs) representing adult (male and female) [1] and children in different ages (newborn, 1, 5, 10, and 15 years male and female) [2] for the calculation of the reference dose coefficients for radiological protection purposes. While providing anatomically more realistic representations than the previous stylized phantoms, the VRCPs have limitations coming from the finite voxel resolutions and inherent nature of voxel geometry. The VRCPs do not precisely represent very small or thin organs/tissues which are below their voxel resolutions (= hundreds of micrometers to several millimeters), resulting in unreliable dose calculations especially for weakly penetrating radiations. Moreover, the VRCPs are not suitable to calculate doses for emergency exposure situations planned by the ICRP, because they are very difficult to be deformed into different postures and body sizes according to the exposed individuals. To overcome the limitations of the VRCPs, the ICRP established Task Group 103 to develop new adult and pediatric mesh-type reference computational phantoms (MRCPs) in a high-quality/fidelity mesh format based on the VRCPs. The Task Group recently completed developing the adult MRCPs and their high fidelity and deformability were demonstrated through various application studies. After development of the adult MRCPs, the Task Group has undertaken to develop the pediatric MRCPs, which are expected to have same advantages as those of the adult MRCPs. The development has almost been completed (see figure 1) and is planned to be fully completed by the end of 2019. This talk will overview the phantom development project in general, explaining the development procedures, main features, and dosimetric impacts.

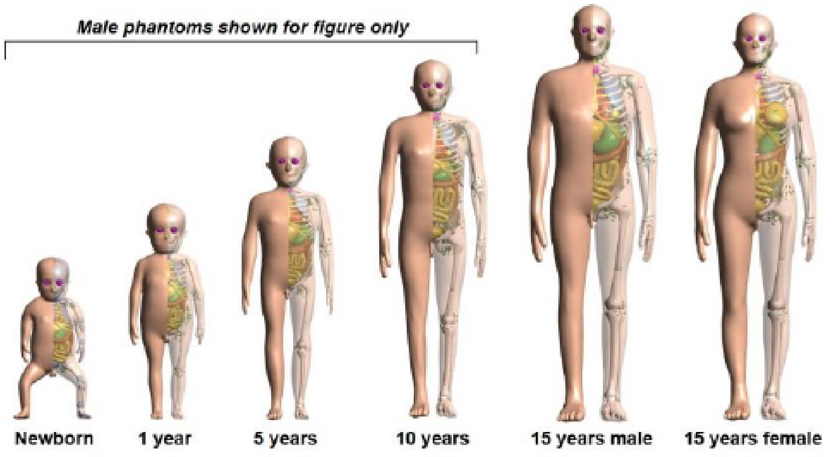

Fig. 1. Preliminary results for development of pediatric MRCPs.

Keywords: ICRP, Pediatric phantoms, Mesh phantoms

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OS2.7 (T2.3-0279)

A Novel Approach of Gamma Spectrum Anomaly Detection Base on Deep Convolutional Neural Network

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Gamma spectrum analysis is a basic technology in RP. At present, the most commonly used analysis method is based on photonelectric peaks in the spectrum. However, when detector's energy resolution is poor, measuring time is short, the spectrum often presents bad behaviours of high statistical fluctuation and low S/N ratio, meanwhile, there will also exist channel drift caused by temperature change. All these behaviours will make misidentification and missing detection very common when the method based on photonelectric peak is applied. How to improve analysis accuracy of these spectra has always been a difficult problem.Recently, pattern recognition technology based on deep neural networks(DNN) has made rapid development. Because of its super performance in feature extraction and classification tasks, it has been widely used in face recognition, image classification, and other fields. Enlightened by this, is it possible to extract the morphological characteristic of gamma-ray spectrum by using DNN and make classification between different categories e.g. background/non-background? In this paper, this idea has been studied.

Firstly, the simulation experiment is carried out. In the simulation, 104 background spectra and 104 spectra containing different amount of 661keV gamma ray (emitted by 137Cs) were generated. Next, these spectra are preprocessed. Then deep convolutional neural network (DCNN) is selected as the learner. DCNN is chosen because of its strong ability to extract morphological features, and more important, it is topologically invariable which means it is resistant to the influence of shape expansion and deformation which are very common in the measured spectra. In operation, we use Tensorflow 1.12.0 to build a DCNN structure; the initial convolution kernel width is 20, and then gradually narrows. The pooling step size is 4, so that the spectrum dimension can be reduced to 1/4 after one pooling; the activation function takes the Relu function; and the optimization algorithm uses SGD method. ROC and detection limit were analyzed. The results show that ROC are better than that of traditional method (using Genie 2000 as a tool), and detection limit are 40% lower than classical Currie limit.Application research was also carried out. A 137Cs point source was measured with a 3 x 3 inch NaI (Tl) detector. The source intensity was changed by moving the position of the point source. 104 background and non-background spectra were measured respectively. The measurement time was 30 seconds. Using these data, ROC and detection limit are also tested, and the results are similar to those of simulation experiment.

The results of this paper show that under the conditions of short measurement time and poor energy resolution, DCNN based detection technology can significantly improve the accuracy and sensitivity of anomaly detection. This conclusion shows that this novel approach has great application potential.

Keywords: *gamma spectrum analysis, convolution neural network, anomaly detection*

OS2.7 (T2.1-0351)

Development of TET2DICOM program for conversion of tetrahedral-mesh phantoms to clinical DICOM-RT dataset

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The computational human phantoms (CHPs) which have been developed in the radiation protection field can be utilized in the radiation oncology, but it is necessary to convert from phantom structure to DICOM-RT dataset [1]. Recently, tetrahedral-mesh (TM) phantoms have been released for representing complex organ structures realistically and anatomically; however, the TM phantoms have never attempted to convert into DICOM-RT dataset. The objective of this research project is to develop the conversion program from the TM phantoms to DICOM-RT dataset, named TET2DICOM, for utilization of the TM phantoms in commercial clinical software. TET2DICOM is written in C++ and contains (1) a new voxelized method to mimic a CT image-like pixel array, (2) a contouring system of organ shape based on VTK library, and (3) exporting DICOM-RT dataset based on GDCM library. To test the developed program, various TM phantoms were converted DICOM-RT format using TET2DICOM. Subsequently, the converted files were imported into clinical software (i.e., *MIM* and *RayStation*) and were checked whether functionalities of software were well performed or not. Our results show that the converted phantoms by TET2DICOM provide identical organ shapes and HU densities within statistical uncertainty of DICOM voxel resolution, and they were successfully implemented in clinical software. Figure 1 shows (a) a pair of TM-type reference Korean phantoms (MRKP-AM and MRKP-AF) [2] and (b) screenshot of planned MRKP-AF converted to DICOM-RT format by TET2DICOM using *RayStation 5,* for instance. More detailed results for the other conversion results, including ICRP reference and pediatric phantoms, will be presented at IRPA15 conference presentation.

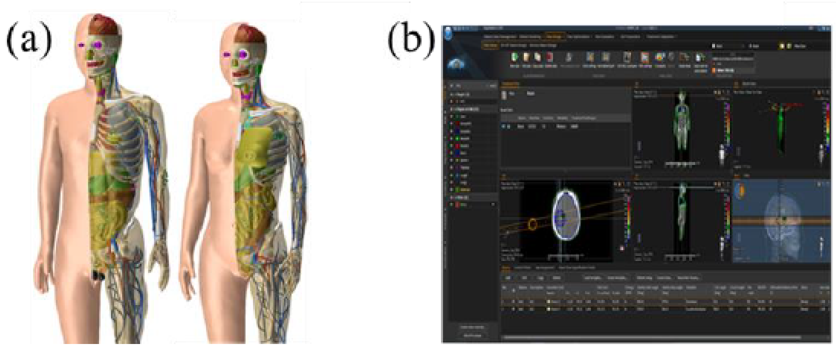

Fig 1. (a) 3D renderings of a pair of TM-type reference Korean phantoms (MRKP-AM and MRKP-AF) [2] and (b) screenshot of planned MRKP-AF converted to DICOM-RT format using *RayStation 5*

Keywords: Tetrahedral-mesh phantom, DICOM, Treatment planning system

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OS2.7 (T2.2-0383)

The influence of using Mesh-type Reference Computational Phantoms on the calibration of an internal exposure measurement device

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There are several nuclear scenarios that represent a threat to the public health and safety. These include terrorism acts or accidents where there is the release of nuclear material. While there are several instruments to quickly measure external exposures, such as Thermoluminescent Dosimeters or Optically Stimulated Luminescence dosimeters, there is not an efficient and fast method to measure internal exposures.

The newly developed mesh-type reference computational phantoms (MRCPs) [1] represent the evolution of the previous reference phantoms and a more detailed description of the human body, addressing the voxel reference phantoms limitations. These allow for a more accurate dose calculation in the human body which in some cases results in a significant difference on the calculated quantities [2].

In this work, the computational calibration of a WBC was performed using the ICRP reference Golem voxel phantom and the male MRCP. With it, it was possible to derive the relationship between WBC measurements and the corresponding effective dose in a person due to internal exposure, which can be used to estimate effective doses to the human body due to internal exposures in emergency scenarios. Whole Body Counters (WBCs) are used to detect and quantify internally deposited radionuclides in the human body and are a standard for in vivo measurements in several internal dosimetry laboratories [3, 4].
The accuracy of the in vivo measurement depends on the anatomical similarity between the phantom used for the calibration and the human body. By assuring this, the isotopes distribution within the body and the corresponding attenuation properties are correctly modeled [4]. Furthermore, it is necessary to precisely quantify the radioactivity deposited by ingestion, inhalation and absorption through the skin. The comparison between the two types of reference phantoms aims at understanding how the more detailed model of the human body provided by the MRCP influences the biodistribution description of the radionuclides and therefore the calculated internal deposited dose.

Measurements were performed using a physical plastic phantom with ^{133}Ba, ^{137}Cs, ^{60}Co and ^{25}Mn sources placed inside in such way that simulate the thyroid, GI tract, lungs and whole body internal deposition. The phantom was placed in an ACCUSCAN II WBC which measurements were performed with two High-Purity Germanium (HPGe) detectors. After validating the corresponding PHITS Monte Carlo model, the computational phantoms were modeled inside the WBC with the organs set as internal radiation sources, to simulate an internal exposure. The simulated source distribution in the organs follows the recommended biokinetic models, for an accurate description of the internal exposure.
This study shows that by using the relationship function obtained from the calibration, the effective dose received to the human body can be derived directly from WBC measurements, making it easy and fast to evaluate the effective dose to the human body by internal exposure in a radiation emergency. The advantages and disadvantages of using the new mesh-type reference computational phantoms is discussed.

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OS2.7 (T2.2-0328)

Assessment of Internal Exposure by Implementing Computational Human Phantoms on the Calculation of the Whole Body Counter Response

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In this study, we attempted to establish a methodology to estimate the internal exposure of the human body from the whole body counter measurement result. The whole body counter is often calibrated with a reference physical phantom such as RMC-II phantom or BOMAB phantom. However, previous studies showed that there is significant discrepancy between the results depending on the phantom used [1], and, furthermore, physical phantoms have inherent limitations on its versatility as they cannot closely reflect the various configurations of the internal exposure. That is, even if the whole body counter is calibrated to the reference phantom, it does not necessarily allow us to correctly estimate the effective dose to the human body from the internal exposure.

We performed various simulations by implementing a few kinds of computational phantoms to be measured in a whole body counter. The human phantoms may include different internal exposure scenarios in respiratory or gastrointestinal organs, or glands. We, first, validated our model for an ACCUSCAN II whole body counter by comparing Monte Carlo simulation results with measurements using a standard point source contained in an RMC-II phantom, and calculated expected responses of the whole body counter to computational phantoms including radioactive sources in organs. Here, we implemented an ICRP reference golem voxel phantom and a HDRK-Man (High-Definition Reference Korean-Man) voxel phantom [2] and calculated the effective dose for the whole body and each organ from the simulation. Various cases and scenarios were simulated to investigate the sensitivity of the whole body counter response to many variables in the assessment of the internal exposure and to obtain a reliable reference data set. The relationship between the whole body counter response and the estimated effective dose for the human body was deduced for several internal exposure scenarios. This methodology can be utilized for various applications requiring internal dose assessment, such as emergency situations including terrorism and accidents.

Keywords: *Internal exposure, Dose assessment, Monte Carlo simulation*

ACKNOWLEDGMENTS

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OS2.8 (T2.1-0208)

Radiation Measurement and Imaging Using Single Pixelated CZT Detector

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The cadmium zinc telluride (CZT) detector has several advantages including high detection efficiency and good energy resolution at room temperature without cooling device due to its high atomic number, high density and wide band-gap. In many studies, the pixelated anode structure had been proposed to achieve the single-charge-carrier sensing and minimize the position dependency of the output signal caused by the charge trapping, low mobility and short life time of the holes, etc. Since this detector also can detect the 3D position where interaction occurs, the correction of the signal loss based on its position, and Compton imaging with a 4π solid angle with the single detector were possible.

In this study, the RENA mini™-long detector system from Kromek Inc. (Durham, UK) which comprised the pixelated 20×20×5 mm^3 CZT crystal, analogy signal integrated circuit and data acquisition board was used. We evaluated its spectroscopic performance and feasibility as a Compton imager for various isotopes. The point sources of ^{57}Co, ^{133}Ba, ^{22}Na and ^{137}Cs were measured and the energy resolutions of 122, 356, 511 and 662 keV peaks were 4.61, 2.94, 2.08 and 2.2 %, respectively. Figure 1 shows the reconstructed Compton images of the multiple sources that were positioned with 30° intervals and simultaneously measured. Each source could be reconstructed separately by using the energy window around each photo peak and all source images were reconstructed by summing individual images after normalization. As a result, every source was clearly identified at each position.

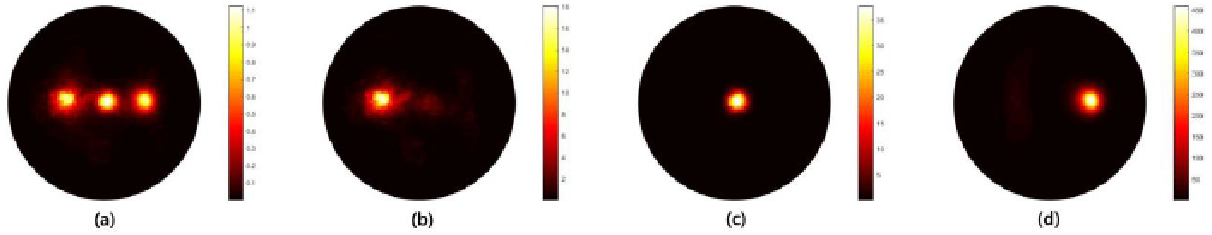

Fig. 1. Reconstruction images of multiple sources
(a) Three sources (b) ^{22}Na (511 keV) (c) ^{137}Cs (662 keV) (d) ^{133}Ba(356 keV)

Keywords: CZT, Spectroscopy, Compton imaging

ACKNOWLEDGMENTS

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OS2.8 (T2.3-0373)

Development of a Novel Method to Perform Clearance Measurements Inside Pipes using PIPS Detectors in a Vacuum

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The decommissioning strategy of Danish Decommissioning is a mixture of decommissioning and decontamination. This means that buildings and structures may remain after decontamination if they can be cleared in their entirety. Pipes and ducts embedded in the structures pose a challenge in the clearance measurements. The chosen clearance measurement strategy for these pipes and ducts is to measure gross α and β using PIPS detectors. As the surfaces of pipes are curved the geometry poses a size constraint of the PIPS detectors, due to the fact that the detector surface needs to be close, within a few cm, to the pipe surface in order to detect α-particles reliably. We would like to present the initial results of a novel measurement method we are developing to mitigate this problem and make more efficient clearance measurements in pipes and ducts.

α-particles have a very limited range in atmospheric air at standard pressure conditions. However this range can be considerably extended in a low density atmosphere. Thus we have manufactured a cylindrical vessel from which we are able to evacuate some of the air and thereby creating a low density atmosphere at ~ 0.02MPa. Placing a plane source and a detector in this environment we are able to do measurements at different source detector distances much greater than the α-particle range at 0.1 MPa. By calculating the geometric efficiency of our detector at the different distances to the source we are able to measure the intrinsic efficiency of our detector at different air pressures. We are calculating the geometric efficiency using a point kernel approach. In principle we should be able to assess the maximum range of the α-particles at different pressures using the ideal gas equation.

Once the intrinsic efficiency of our detector has been determined we will be able to calibrate our measurement setup. Measuring in a low density atmosphere allows a larger detector as the curvature of the pipe and maximum distance from the surface no longer poses a constraint of the size of the detector. We are planning to use multiple detectors in an arrangement which allow us to measure a segment of the pipe in one measurement. In our planned setup a detector is situated perpendicular to a radial line of the pipe at a given distance from the curved surface. The group of detectors is then placed in a polygonal fashion allowing a simultaneous measurement of an entire pipe segment. In order to calibrate our measurement setup we need to calculate the geometric efficiency of the detector measuring on a curved surface. This will be done using the point kernel method. The calculations will also provide information on how many detectors at a given size is needed to measure a section of a given pipe diameter.

In order to control the width of the measured section our plan is to sandwich the detector array between two discs with a diameter corresponding to the internal pipe diameter. The discs also serve as gaskets, which ensure that we are able to evacuate the air between the discs and create a low density measurement atmosphere.

We will present the development of a novel clearance measurement method, to measure embedded pipes and ducts efficiently. We will show preliminary results using a cylindrical vacuum vessel, in which a detector can be positioned at different distances from a plane source. The calibration of this laboratory setup will be used to calibrate the final measurement setup, where multiple detectors are utilized to measure a pipe segment at the same time. It will be shown how the calibration factors are obtained and also show the design of a measurement setup. Finally it will be touched upon how the measurement method will be integrated into an accredited clearance measurement system.

Keywords: Clearance Measurements, PIPS detectors, Pipes

OS2.8 (T2.3-0592)

Development of Alpha Particle Detectors for Detecting Radiological Contamination

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Alpha contaminations in nuclear facilities should be detected instantaneously because alpha particle emitters such as plutonium isotopes are very harmful when a worker inhales them. We have developed new alpha particle detectors for detecting the alpha contaminations accurately and instantaneously: alpha imaging detectors and alpha dust monitors.

An alpha detector which has a good energy resolution and spatial resolution is useful for distinguishing nuclear material such as plutonium from radon progeny. For the development of the alpha imaging detector, we used a thin cerium-doped $Gd_3(Ga,Al)_5O_{12}$ (Ce: GAGG) scintillator and silicon photomultipliers (SiPMs). The thin GAGG scintillator was optically coupled to a light-guide and the SiPMs for the detector fabrication. The detector showed a good energy resolution for 5.5 MeV alpha particles (~13 % at the full width at half maximum (FWHM)). The detector is capable of capturing two-dimensional alpha images. This detector also has the advantage of its compact size, which enables to measure the alpha contaminations in narrow spaces. We demonstrated the actual measurement of smear samples obtained from the Fukushima Daiichi Nuclear Power Station using the developed imaging detector [1].

Commercial alpha dust monitors with a silicon surface barrier detector (SSBD) operating at some nuclear facilities frequently produced false alarms due to environmental conditions such as humidity. For the development of the alpha dust monitor, we used a cerium-doped $Gd_2Si_2O_7$ (GPS) scintillator plate and a photomultiplier tube (PMT) [2]. The energy resolution for 5.5-MeV alpha particles was ~12% FWHM. The count-rate of the radon progeny decreased by 77% with applying energy discrimination. The alpha dust monitor was capable of conducting alpha-particle spectroscopy even though the GPS scintillator got wet. The alpha dust monitor is an ideal choice for in places lacking temperature and humidity controls.

In the presentation at the IRPA 15 conference, we will present the developments and measurement results of these alpha detectors.

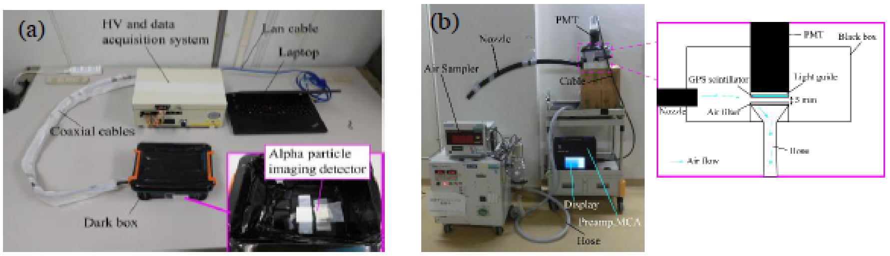

Fig. 1. Developed alpha imaging detector (a)[1] and alpha dust monitor (b)[2]

Keywords: Alpha contamination detector, Alpha imaging, Alpha spectrometry

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OS2.8 (T2.3-0623)

Development of radioactivity measurement using isothermal calorimetric system by comparing with TDCR counter

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The radioactivity measurement method using a calorimeter uses the 1st law of thermodynamics. The decay energy of a nuclide is absorbed in isothermal system (calorimeter) and converted into thermal energy. The converted thermal energy causes a temperature change in the chamber. Heat flow occurs due to temperature difference between sample and background. The heat flow rate of calorimeter is converted into a measurable electrical power by the thermoelectric sensor using the Seebeck effect. Radioactivity can be calculated using the decay energy of the radionuclide, the absorption efficiency at which decay energy is absorbed by the calorimeter, the calibration factors of the system, and the measured heat flow rate.

The isothermal calorimeter measurement method can overcome the problem of counting loss due to dead time. However, since the measurement range of the calorimeter differs from that of the ionizing radiation detector, studies are being conducted to secure traceability of radioactivity.

TAM III (TA Co.) was used as a measuring instrument. By circulating the mineral oil in oil bath, temperature changes inside TAMIII can be minimized and the temperature setting of the system can be maintained.

The calibration factor of the calorimeter was obtained using the joule-heating method. The resistor of 199.978 kΩ connected with current source and voltagemeter was placed in calorimeter. The current of current source was supplied and voltage was measured by DAQ. The expected power (generated from known current and resistance) and the heat flow rate measured in calorimeter were compared to obtain calibration factor. The calibration factor was obtained in the form of linear function such as $y = 1.015\ x + 0.078$.

^{32}P radionuclide was used with average beta energy of 695.5 keV per Bq for radioactivity measurement. The absorber is made for the purpose of 100% absorption of average decay energy. As a result of using Monte Carlo simulation, the decay energy absorbed in the absorber was almost equal to the average beta energy. ^{32}P radioactivity in the calorimeter can be obtained by applying the calibration factor after dividing the measured heat flow rate by 695.5 keV per Bq. The radioactivity measurement of ^{32}P source in calorimeter was about 3 weeks and subsequent radioactivity measurement was replaced by TDCR. In addition, we assume that effect of heat defect on the liquid source will affect the heat flow rate measurement. In order to reduce the possibility of counting loss of calorimeter due to heat defect, the volume of liquid source was adjusted to 3 mL.

As a result, the results of comparing the radioactivity (calorimeter) and the TDCR measurement had a difference of 2.3 %. The result was at the time when the effect of dead time is expected to be large. After a period of time to overcome the effects of dead time, the comparison seems to be good. In addition, it would be possible to measure radioactivity more precisely in the calorimeter if the effect of heat defect on the liquid ratio of liquid source is demonstrated.

Keywords: heat flow rate, absorption efficiency, heat defect

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OS2.8 (T2.3-0225)

Development of Innovative Methods in Radionuclide Metrology for Applications in Natural Resources and Life Sciences

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Naturally occurring radionuclides like ^{40}K and the decay products of the primordial radionuclides ^{232}Th, ^{235}U and ^{238}U are present in many natural resources. Naturally occurring radioactive materials (NORM) containing these radionuclides are exploited industrially and often exceed the exemption limits of the activity concentration. In addition, industrial activities generate a significant portion of waste, possibly enhancing the potential of exposure of workers and the public and the management.

It is necessary to create traceable, accurate, and standardised measurement methods, reference materials and systems for application in the concerned industries and laboratories to ensure correct analysis of activity concentration and ensure the radiation protection of workers and the public. There is a general need for certified NORM reference materials in science and industry to achieve traceability, validate methods and calibrate instruments.

The main problem with NORM lies in the variety of chemical elements and radionuclides composition. NORM emits many gamma-rays of different, and sometimes interfering, energies that have to be measured and analyzed by an expert. For the traceable determination of activity concentrations in NORM, reference materials with a well-established activity concentration are essential.
Therefore, an extensive study of particular problems arising in γ-ray spectrometry of NORM samples has been conducted to investigate the best gamma-lines to analyse the activity concentration in NORM.

A novel alternative way to approach this problem is the use of artificial neural networks (ANNs). ANNs are mathematical software tools that emulate the way the human brain works. They are trained, tested and validated using sample datasets and can generalise the "knowledge" gained from the content of the training set, applying it to new problems. This can be viewed as a new calibration tool where no expert knowledge of gamma-ray spectrometry is needed by the end-user. In this work an ANN was created in the frame of MetroNORM that is able to decide from the input data of a raw gamma-ray spectrum if the activity concentrations in a sample are above or below the exemption limits. Six NORM reference materials have been analysed. To widen the applicability of the algorithm, a set of artificial gamma-ray spectra with varying densities and activity concentrations and material compositions have been created by Monte Carlo simulation and successfully used in the training, testing and validation of the ANN.

Additionally, a certified traceable reference material has been established. It is made from filter sand of a drinking water production site with significantly elevated levels of ^{226}Ra and was used in an intercomparison exercise with nine European laboratories concerned with radioactivity measurements. Due to its volatile nature, special attention is given to the evaluation of radon activity concentration and radon tightness of used sample containers. A simple and sensitive method for the estimation of 222Rn leakage is proposed.

Furthermore, two ionization chamber methods have been used to evaluate the radon activity concentration in drinking water samples.

The results of these studies, the preparation of the reference material and the results of the intercomparison exercise and ANN will be presented in this paper.

Keywords: radionuclide metrology, NORM, ANN

OS2.12 (T2.3-0607)

Activity standardization of ^{134}Cs and ^{137}Cs by 4πβ(LS)-γ coincidence counting method

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Caesium-137 (^{137}Cs) and Caesium-134 (^{134}Cs) are two important radioisotopes that are widely used in both the research and industrial field. In most nuclear accidents, these isotopes are often used as parameters for radiation contamination in marine organisms, foodstuffs, as well as in the environment. The ratio of these two isotopes in the environment was also utilized to identify the reactor units that caused atmospheric releases during the Fukushima nuclear accident [1]. In terms of activity measurement, the two isotopes are very important for the calibration of γ-ray spectrometer as well as for the study of coincidence summing on gamma spectrum analysis. Due to the importance and the wide application of these two isotopes, activity standardization of these isotopes has become important to provide an accurate and traceable activity measurement of the isotopes.

^{137}Cs disintegrates with a half-life of 30.05 (8) years by β-ray emissions. The emissions occur 94.36% to the excited level of ^{137m}Ba with end-point energy of 513.97 (17) keV and 5.46% to the ground state of ^{137}Ba with end-point energy of 1175.63 (17) keV. The meta-stable ^{137m}Ba further decays with a 661.6 keV γ-ray emission to the ground state with decay time 2.552 minutes [2]. This long decay time makes the time correlation between the 513.97 keV β-ray and the 661.6 keV γ-ray emission events nearly completely disappeared. Therefore, for the case of activity measurement, ^{137}Cs needs to be treated as a pure β-ray emitter that makes the conventional 4πβ-γ coincidence method cannot be applied directly to determine its activity. For this reason, ^{134}Cs with a known activity value is used as a tracer to determine the activity of ^{137}Cs through efficiency tracing method. To be used as a tracer, the ^{134}Cs needs to be standardized to obtain its activity value. The standardization is performed by using 4πβ(LS)-γ coincidence counting method. The complex decay scheme of ^{134}Cs suggests the standardization of this nuclide using different γ-window may lead to different efficiency function.

This paper described the activity standardization of ^{134}Cs based on five different settings of γ-windows and the activity standardization of ^{137}Cs using an efficiency tracing method. The mean activity value of ^{134}Cs given by measurements from all γ-windows setting was (1256.0 ± 3.2) kBq/g on reference time with activity value obtained from each window differing by a maximum of 0.46%. The final activity value for ^{137}Cs obtained from efficiency tracing method time is found to be (942.0 ± 5.8) kBq/g on the same reference time. All quoted uncertainty values are evaluated at $k = 1$. All final activity value was compared to the measurement with the standard ionization chamber.

Keywords: 4πβ-γ coincidence method, ^{134}Cs ^{137}Cs activity standardization, efficiency tracing method

ACKNOWLEDGMENTS

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OS2.8 (T2.3-0223)

The MetroRADON Project – progress and findings after 3 years European research work

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MetroRADON is a 3-year Horizon 2020 Joint Research Project funded under European Metrology Programme for Innovation and Research (EMPIR) in which 17 European metrology and research institutes aim to provide metrology for radon monitoring. During the project, reliable techniques and methodologies to enable SI traceable radon activity concentration measurements and calibrations at low radon concentrations have been developed. This will help to establish a basic European metrological infrastructure for radon measurements, enabling sound and traceable monitoring of radon and radon protection in Europe. The need for this project is mostly motivated by the requirements of the implementation of the European Council Directive 2013/59/EURATOM (EU-BSS), aiming to reduce the risk of lung cancer for European citizens due to high radon concentrations in indoor air.
Uptake and exploitation of the project's results and dissemination to as well as inclusion of experiences by all stakeholders concerned with radon was another main focus point that will be continued by the partners after the end of the project. The methods developed in the project will assist EU Member States in the establishment of their national radon action plan required under the EU-BSS. The development of a novel European unified index of geogenic radon hazards provides a consistent picture of susceptibility to geogenic radon across Europe and will be an important tool for the harmonised implementation and performance of national radon action plans. Novel calibration methods and traceability validation at low radon activity concentrations have been devised and new and stable radioactive reference sources developed to achieve sufficiently low uncertainties. For the first time, the distortion of radon measurement results due to the presence of thoron is considered and corrected at low radon activity concentrations. Traceability to a primary thoron standard is ensured (and has been refined), enabling the thoron influence to be reliably investigated. Several materials and methods that can serve as thoron barriers have investigated. Several intercomparisons have been conducted in the framework of the project, amongst others a Consultative Committee of Ionising Radiation CCRI(II) key comparison at the International Bureau of Weights and Measures (BIPM). These intercomparisons serve to strengthen the trust of stakeholders and end-users in the measurement and calibration capabilities of the participating laboratories.
Guidelines and recommendations on the new calibration and measurement procedures have been and continue to be published and made available to end users, standards organisations, regulators and international bodies. This paper will present the main results and research highlights after the end of the 3-year Joint Research Project MetroRADON.

Keywords: Radon, metrology

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OS3.1 (T3.1-0215)

Personnel Radiation Monitoring Lessons Learned – A Dose Reconstruction Perspective

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Under the U.S. Energy Employee's Occupational Illness Compensation Program Act (EEOICPA) of 2000, the National Institute for Occupational Safety and Health (NIOSH) is responsible for reconstructing worker radiation doses that are subsequently used to determine the probability that a worker's exposure caused a specified cancer. This work discusses personnel monitoring lessons learned from a dose reconstruction and radiation compensation perspective. Personnel monitoring for both internal and external radiation exposures has changed many times over the past 80 years of radiological monitoring. Most changes have been good and improved radiation monitoring and subsequent radiation protection. For example, most radiological workers are monitored for radiation exposure via a personal dosimeter and/or through biological monitoring when there is an internal exposure potential. However, some changes such as decreasing or eliminating personnel monitoring in favor of area, group, or co-worker monitoring have led to challenges from a dose reconstruction and radiation compensation perspective. The scaling back of personnel monitoring whether external dosimetry or bioassay monitoring has resulted in the need to rely on documentation of the worker's radiation environment. Without this documentation, dose reconstruction can be challenging when a worker presents with a reported injury (cancer) and the job of the radiation safety professional is to reconstruct the worker's radiation dose. Was the worker ever exposed to radiation? Was the worker not monitored because the expected dose was less than the personnel monitoring requirements? How does the radiation safety professional demonstrate this? Were the area or co-worker records maintained and accessible? Are compliance records sufficient? Were there breakdowns in the radiological monitoring that led to non-compliance issues? How does the radiation safety professional reconstruct the dose in these scenarios? When personnel monitoring data is not available, alternate radiation monitoring data (air sample, surface contamination, co-worker data) becomes critical for dose reconstruction and compensation decisions. This presentation provides examples of radiation monitoring lessons learned over the past 18 years of dose reconstruction under EEOICPA. The lessons learned include availability and subsequent analysis of personal and alternate radiation monitoring data. The presentation will also address monitoring data limitations such as detection limits (missed dose), records quality and availability, as well as difficulties placing workers in different radiation environments during their career and how these factors can impact the ability of the radiation safety professional to reconstruct a worker's dose.

Keywords: Dosimetry, Bioassay, Dose Reconstruction

OS3.1 (T3.1-0047)

Application of the "graded approach" for the radiation protection of workers – Examples and reflexions from European ALARA networks

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The "graded approach" to the regulatory control of practices and the issuing of authorisation is a key principle in the Euratom Directive 2013/59 [1] that states that the approach applies "*for the purpose of radiation protection and* [shall be] *commensurate with the magnitude and likelihood of exposure*" (art. 24). Exemption, notification, registration/licensing and authorization as well as the way to perform inspections (administrative, technical, etc.) are the options in the hands of the regulator to apply the graded approach (art. 25 to 30). The Directive had to be implemented into national legislations by 2018.
In this context, the European ALARA Network (EAN[1]) and the European Radiation Protection Authority Network (ERPAN[2]) has decided to set up a brainstorming meeting in December 2018 to discuss about the interpretation of the graded approach and also its practical implementation.
The scope of the brainstorming has been limited to the application of the graded approach for workers at their workplace. To animate the debate, the participants (from 7 countries) elaborated several keynotes to cover different sectors of activities. In addition, several questions have been shared prior to the meeting, *inter alia*:

- What are the factors used to implement the graded approach? Are these factors only linked with radiation protection?
- Are there differences in the graded approach for artificial sources and 'natural' sources like radon, or NORM?
- Can we identify good practices and factors of success, or conversely difficulties, when implementing a graded approach?

After the meeting, the interest raised by the discussions was elevated and the participants decided to launch a survey via the EAN and ERPAN networks. The survey's objectives were to collect additional points of view and also more examples of application. The survey ran in Spring 2019 and 13 questionnaires were sent back, by regulators and professionals.
A synthesis of the brainstorming meeting as well as the results of the survey will be presented. Several themes will be addressed like the objectives of a graded approach, notably from the regulators point of view; the common and transversal aspects across the countries/sectors when shaping a graded approach and also the existence of differences and the reason of it. If the 'entry point' of the approach is generally indeed based on the "*magnitude and likelihood of exposure*", it also appeared that a wide panel of factors – and not only radiological ones – are used in the grading it-self, depending on the sector and the circumstances. Examples of good practices that support the implementation of a functional graded approach have been collected and will be presented.

Keywords: Graded Approach, Euratom Directive Basic Safety Standards, ALARA

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OS3.1 (T3.1-0406)

QUANTITATIVE EVALUATION OF RADIATION PROTECTION IN INDUSTRIAL GAMMA RADIOGRAPHY THROUGH REGULATORY INSPECTION

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Industrial gamma radiography is one of the industrial activities deemed as high radiological risk. It is classified by IAEA as Category 2, among five classes, due to the high probability of occurrence of radiological accidents and the use of high radioactive activity sources. Industrial radiography is responsible of relatively significant number of accidents in the world. These conditions have been receiving special attention from the Regulatory Authorities to maintain a high radiation protection level in radiography installations. This work presents a methodology to evaluate quantitatively the radiation protection in industrial gamma radiography installations during regulatory inspection. It was based on five steps: (1) analysis of 40 radiological accidents cases published by IAEA; (2) link among these accident cases, the main causes and the requirements items for regulatory inspection; (3) selection of 25 requirements items from step 2 to link with IAEA enforcement classification; (4) definition of weight value for each enforcement group depending on the risk of non-compliance; (5) verification, at the end of regulatory inspection, the negative points received by installation due to its non-compliance, for a total of 390 points. The methodology used showed that the higher score received by the installation means that the radiation protection level is in risk. The quantitative evaluation done after a regulatory inspection is a resource to verify the performance and specially to quantify of radiation protection level based on non-conformities. It can also become a tool of great assistance in the licensing process runs by the Regulatory Authority.

1. INTRODUCTION

Industrial radiography is one of the non-destructive inspection methods widely used in the inspection of ferrous and non-ferrous materials welds, castings and forgings, where quality requirements of the industrial sector are necessary to avoid discontinuities in parts, components, equipment, etc. This method uses radioactive sources of ^{60}Co, ^{75}Se and especially ^{192}Ir [1].

As industrial gamma radiography is one of the industrial activities deemed as high radiological risk it is classified by IAEA as Category 2, among five classes, due to the high probability of occurrence of radiological accidents and the use of high radioactive activity sources [2].

The work involved is often carried out under difficult working conditions, such as inside confined spaces, in extreme cold or hot temperature, or during the night. Working under such adverse conditions might result in operational situations in which occupational radiation protection procedures may be compromised. Experience shows that incidents involving industrial radiography sources have sometimes resulted in high doses to workers, causing severe health consequences such as radiation burns and, in a few cases, death [3].

Industrial radiography is responsible of relatively significant number of accidents in the world. In the last 30 years, more than 80 severe radiological accidents involving 120 radiation workers, 110 members of the public and 12 deaths happened in the world, including 10 accidents in Latin America [4, 5, 6, 7]. IAEA published three Latin America reports: Peru (1999), Bolivia (2002) and Chile (2005).

These conditions have been receiving special attention from the Regulatory Authorities to maintain a high radiation protection level in industrial installations. In order to control the use of radioactive sources, specifically in industrial radiography, all countries should establish a Regulatory Authority with very specific functions, taking into account IAEA recommendations. National Nuclear Energy Commission (CNEN) is the Brazilian Regulatory Authority and in Mozambique is the National Atomic Energy Agency (ANEA). Both

OS3.1 (T3.1-0655)

Radiation safety standards for public and workers protection in Belarus

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International requirements have been introduce to national radiation safety legislation and requirements of the Republic of Belarus since 2012 for maintaining national system on radiation safety and one of triggers was decision to build new Belorussian NPP. The Republic of Belarus is constructing the first NPP with two units of WWR-1200, 1st and 2nd NPP Units are planned to commence operation in 2020.

The Ministry of Health, as one of the radiation safety regulators, is responsible for requirements process, inspections, dose assessment for public and collects the data of radioactive monitoring in people locations, radioactive contamination of commodities (foodstuff, drinking water), workers and public dose assessment and etc.

The revised Law on Radiation safety (2019) and national safety requirements for public and workers protection (SANPINs) took into account the requirements of IAEA Safety Standard GSR Part 3 regarding the separate consideration of the situations of planned, existing and emergency exposure. For each situation, the appropriate approaches for limitation and optimization of the public and staff exposure have been established: for the planned exposure situation - Dose Limits and Dose Constraints (DC), for emergency and existing exposure - Reference Levels (RL) of exposure doses to the public, reference levels of content of radionuclides in the environment, foodstuffs and products. The Requirements define DC for practice, for example for newly constructed NPP as 100 µSv/year. Specified DC for radioactive waste disposal facility is 300 µSv/year.

The general emergency requirements, generic criteria for implementing protective actions in case of radiological or nuclear emergency, emergency zones and distances have been established are compliance with IAEA Safety Standard GSR Part 7. Further these criteria have been implemented in the National Emergency Response Plan and make a basis for decision making on taking protective actions in case of radiological or nuclear emergency. In addition, they were used to perform a hazard assessment of nuclear facilities and determine their Emergency Preparedness (EP) category in line with Requirement 4 of IAEA Safety Standard No GSR Part 7. The categorization is to facilitate introducing of graded approach to applying EP requirements in the regulatory system of the Republic of Belarus.

The Law On Radiation Safety (2019) established new legislative basis accordance with IAEA requirements and prescribes using of graded approach to application of radiation safety requirements for supervision of radiation facilities and emergency preparedness measures. This means that state supervision over radiation facilities should be carried out taking into account the degree of radiation hazard of the facility and its risk to the health and life of personnel and the public.

An important advance was the introduction to the requirements the terminology of ICRP and GSR Part 3: the term "quota" for "dose constraint", "representative person" for "critical group", "minimal activity" for "exemption and clearance levels" and etc. There is not only a new terminology but for the first of all there are new approaches on safety regulation and assessment.

The Belarusian experience on legislation processing since 2012 allowed to establish gaps and the needs for new regulatory requirements or recommendations and can be beneficial for radiation safety requirement process in other countries.

The new radiation protection approaches can be intentionally and deliberately introduced into national requirements, for example: representative person safety conception, justification principals introduction using reference levels and dose constrains, clearance and exclusion of radioactive material from regulations and control, new radiation waste classification and management and etc. The gaps and weaknesses that need to be considered when implementing international recommendations to national legislation have been determined and recommendations that have to be work out to comply with new requirements were identified as a results many years experience of processing and use of new radiation safety requirements in Belarus.

Keywords: Radiation safety, requirements, NPP, radiation protection, public, workers.

OS3.1 (T3.7-0630)

An Alternative Proposal for the Regulatory Framework of the Large Accelerator Facilities in Korean Nuclear Safety Act

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Recently, the large accelerator facilities such as PAL-XFEL, Pohang Accelerator Laboratory hard X-ray free electron laser, and RAON, a heavy ion accelerator of Institute for Basic Science, as well as the particle accelerators for radiation therapy have been introduced and are being constructed continuously in Republic of Korea. Although the domestic trend is increasing, the regulatory criteria, suitable for the small radiation generating devices, are applied to the large accelerator facilities. As the large accelerator facilities need to be regulated by the concept of the graded approaches to construct, the regulatory framework changes and improvements in Korean Nuclear Safety Act are required to review the radiation safety issues of the large accelerator facilities.

To evaluate the alternative proposal, we reviewed and investigated the status of the acts, the regulations, the classification and the licensing procedures for the large accelerator facilities in foreign countries. We also considered the accelerated beam specification of the current and near-future domestic large accelerators and the results of the assessment of the radiation source, i.e. neutron yields and the induced radioactivity by the Monte Carlo code, FLUKA. As the results, we proposed two classifications for the large accelerator facilities according to the radiation risk and summarized the regulatory requirements according to the classification, level 1 and 2. It is considered appropriate to classify the licensing procedures into two steps, the construction permission and the operation permission, and related legislation amendments of the domestic large accelerator facilities are proposed. The results of this study would help to develop the regulatory framework of the large accelerator facilities in Korean Nuclear Safety Act.

Keywords: Large Accelerator Facilities, Licensing Procedure, Regulatory Framework

ACKNOWLEDGMENTS
We wish to thank the experts in Accelerator Radiation Safety Forum (ARSF) in Korea for their review of the results. This work was supported by the Korea Institute of Nuclear Safety (Grant No. 19-22).

OS3.1 (T3.1-0386)

Operational indications and self-assessment forms regarding environments dedicated to the handling of unsealed sources for medical purposes

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Nowadays the use of radionuclides in the health sector is increasingly widespread in nuclear medicine applications, or in activities concerning the manufacture and use of radioactive sources for preparing radiopharmaceutical products.
In Italy a comprehensive guide for designing medical laboratories where unsealed sources are handled has been issued in collaboration with one of the national regulatory authorities. In that document indications are provided to fulfill the Italian regulation and to guarantee the workers safety.
In this work we intend to propose the most important suggestions reported in the published guide, providing useful operational indications to approach, with the best practice, the design of a "complex" nuclear medicine service. Special attention is devoted to organizational and safety aspects.
Main goal is to provide an essential guide in evaluating and choosing the most suitable features and equipment to limit the risks due to ionizing radiations and to prevent contamination of workers and the environment.
In order to prompt toward a sistematic application of the proposed indications, the article includes specific self-assessment forms, consisting in a list of some different sequential items, which address the design process in all the different phases.
The approach followed in the elaboration of the forms includes a list of key points of the design process and four levels of applicability for every single indication.
These forms represent a useful tool-box both in the initial phase of the project and subsequently for the development of internal audits aimed at assessing compliance with the indications provided.

Keywords: indications, self assestament forms, Radiopharmacy, Nuclear medicine, unsealed sources

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Oral Session T3

OS3.1 (T3.7-0172)

Radiation safety issues regarding X ray emittable devices below 10 kV applied voltage

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The exemption level for radiation generators or an electronic tube is (i) They do not in normal operating conditions cause an ambient dose equivalent rate or a directional dose equivalent rate, as appropriate, exceeding 1 μSv/h at a distance of 0.1 m from any accessible surface of the equipment; or(ii) The maximum energy of the radiation generated is no greater than 5 keV according to IAEA GSR Part 3 and RSG1.7.

The purpose of this presentation is to propose a method to solve issues in terms of radiation safety issues regarding x-ray emittable devices below 10 kV applied voltage and the literature review was performed.

In Japan, except national government, Industrial Safety and Health Act covers X ray equipment and it defines the specific X-ray equipment which are required to use irradiation tubes and filters, and to limit the irradiation field adequately. However, the exemption for x-ray equipment has not been implemented in Japan yet as of Sep. 2019. Therefore, there are practical differences of radiation safety protocol in each facility on X ray emittable devices such as for X ray fluorescence analysis and electronic microscopes. For distribution of the newly developed basis weight gauge with below 5 kV X ray generator, it is expected to implement the international exemption level to the Japanese regulation. Furthermore, Crookes tubes including old fashion types are popular in junior high schools and the updated government course guidelines mentioning Crookes tubes will be fully implemented in 2021 so that teachers require radiation safety information since it is difficult to measure low energy and pulsed X ray in a junior high school. Therefore, a unique project for providing accurate radiation doses around each Crookes tube in a school is ongoing in Japan.

Although it is not explicitly mentioned in GSR Part 3 or RSG1.7, the recommended reference depths would be 10 mm for the operational quantity ambient dose equivalent and 0.07 mm for directional dose equivalent, since for low energy photons, the reference depth should be 3mm or 0.07 mm for directional dose equivalent that should be evaluated according to Para 2.34 to 2.38 of GSG7. However, exemption level was derived by using a scenario to keep each effective dose of 1 mSv in a year not mentioning an equivalent dose to the lens of the eye of 15 mSv nor an equivalent dose to the skin of 50 mSv in a year.

For this reason, it was thought that the exemption level indicated by the IAEA for X ray equipment should be implemented in Japan and the criteria regarding depth on an ambient dose equivalent rate or a directional dose equivalent rate for exemption should be clearly defined depending on photon energy. Furthermore, dose constraints for teachers and junior high school students should be established by stakeholder involvements.

Keywords: *Exemption, X ray emittable devices, dose constraints for students*

ACKNOWLEDGMENTS

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OS3.2 (T3.1-0455)

The role of clearance and exemption in regulating radioactive liquids using a graded approach: A UK perspective

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Liquids wastes containing radioactive substances may be generated by a wide range of facilities and activities, e.g. the decommissioning and clean-up of legacy nuclear sites, production water from the oil and gas industry, waste waters resulting from medical applications of radionuclides. The regulation of those radioactive liquids should be in accordance with the graded approach.

The concepts of clearance and exemption are important elements of the regulatory framework which, if applied appropriately, contribute to the application of a graded approach, whilst maintaining a high level of protection. The IAEA have provided guidance on the application of the concepts of clearance and exemption. However, that guidance does not explicitly address liquids. Also, unlike for solids, there are no “bulk quantity” values provided for liquids. It is also questionable whether the moderate quantity values address all liquid exposure pathways.

The UK has recently reviewed the regulatory framework for controlling the management of radioactive liquids. The review consisted of 4 tasks; (1) a review of relevant international standards and guidance, (2) a collection of information, via questionnaire, from other countries regarding how they apply the concepts of clearance and exemption to liquids in their own regulatory framework, (3) a review of the UK’s domestic framework and its implementation using real examples, and (4) recommendations for improvement.

Twenty countries responded to the questionnaire; the responses indicate that there is no consistent approach to the application of clearance and exemption to liquids. This reflects our finding that there is relatively little international guidance on this topic.

Real examples of liquid waste management in the UK indicate that more careful application of the concepts of clearance and exemption will improve the regulatory framework and result in a more graded approach.

OS3.2 (T3.1-0339)

ALARA in Practice – 4 Decades of Radiological Protection at Goesgen NPP

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ALARA is an important tool in protecting people from the harmful effects of ionizing radiation. It supports both, regulators and operators in implementing the three radiological protection principles [1], namely limitation, justification and especially optimization. Even though these principles have not been in the foreground in the early years of nuclear power generation, operators initiated programs to optimize both collective and individual doses to workers and the public. Due to technical progress and growing experience, these optimization programs resulted worldwide in a significant reduction of occupational exposures to workers in nuclear facilities. At Goesgen NPP as one example mean individual doses dropped from 3.5 mSv/y in the mid 80ies down to less than 0.3 mSv/y today. Similar reduction factors were obtained regarding maximum individual and collective doses. Even though reduction gradients decreased over time [2], the downward trends are continuing.

Because of these good results as a consequence of our efforts and the global cultural change on how society is facing any kind of risk the initial ALARA philosophy is moving more and more to a continuing expectation that optimization is equivalent to reduction. This philosophy change is even more pronounced in the light of the ongoing discussions about the effects of low dose radiation. Multiple conservatisms in dose models and dose calculations as well as added conservatism due to uncertainties will lead to a misbalance between radiation risks and benefits. That causes anxiety among the public and an unnecessary economic burden to nuclear facilities. In light of pressing global issues (i.e. climate change), a discussion about the meaning of the 'R' in ALARA is urgently recommended.

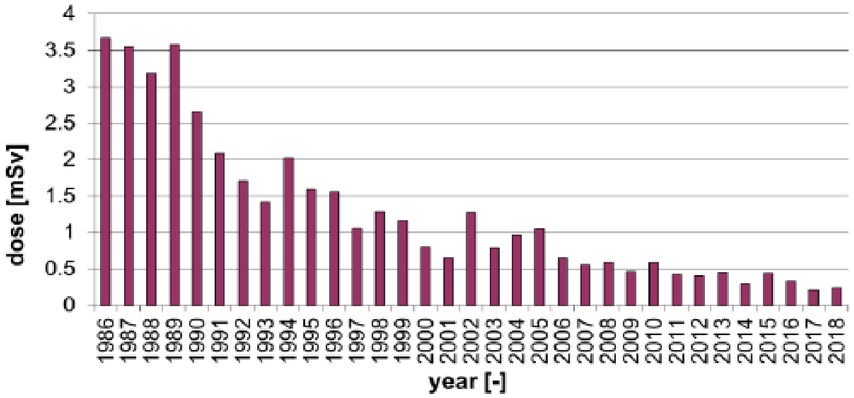

Fig. 1. Mean individual doses at Goesgen NPP

Keywords: ALARA, Optimization, NPP

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OS3.2 (T3.1-0197)

Issues to be Discussed Regarding Application of Conformity Assessment on Uncertainty of Measurement to Radiological Safety Regulations

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JCGM106 (ISO/IEC Guide 98-4: 2012)[1] is an internationally used document providing standards or guidelines regarding conformity assessment considering the uncertainty of measurements. This guide provides general guidance and procedures for assessing the conformity of an item (entity, object or system). The Joint Committee for Guides in Metrology (JCGM) consists of eight organizations including the International Organization for Standardization (ISO) and the International Electrotechnical Commission (IEC), and has not been directly linked with the radiological protection community. The scope of this guide seems to be limited to the application to product control with a very severe requirement for accuracy, for example, in the size or mass of the products. However, in reference to JCGM106, the German Commission on Radiological Protection (SSK) adopted a recommendation[2] regarding the conformity assessment in radiation measurement in September 2016. An example of the conformity assessment in this recommendation is shown in Fig. 1.

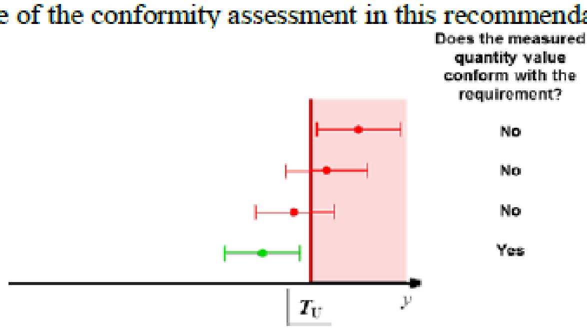

Fig. 1. Example of conformity assessment, where T_U is the regulatory level, in German recommendation[2].

Figure 1 shows that the upper confidence level of the measurement must be below the regulatory level, taking relevant uncertainties into account. This requirement of conformity assessment was also incorporated into a Japanese regulatory guide in compliance with the clearance level by the Nuclear Regulatory Authority (NRA) in September 2019[3]. Also, since 2018, the need for this requirement has also been discussed at an international level in the framework of the ongoing revision of the IAEA Safety Guide RS-G-1.7. In this presentation, whether such a strict requirement of uncertainty is necessary for radiological safety regulations is reviewed from various viewpoints, e.g., the methodology used to derive clearance levels, the balance in a radiological protection system with a graded approach, the difference between product control and radiological protection, and the understanding of the health risk of radiation by both the public and the regulators.

Keywords: Uncertainty, Conformity assessment, Regulation

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OS3.2 (T3.1-0060)

Psychophysiological examinations of employees of the nuclear industry - an element of the radiation protection system

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The relationship between the level of health of the employee and his professional reliability at the moment is obvious and requires no further proof. The implementation of measures aimed at reducing the risks of emergency situations caused by the human factor at nuclear power facilities is an important element of the radiation protection system.

Annual psychophysiological examinations of certain groups of employees of the nuclear industry in the Russian Federation is a prerequisite for admission to work. These surveys allow timely detection of violations of adaptation, leading to a decrease in professional reliability, and to carry out targeted rehabilitation measures.

Every year in State Research Center – Burnasyan Federal Medical Biophysical Center of Federal Medical Biological Agency conducts up to 500 surveys of employees of the nuclear industry working under the influence of ionizing from 3.25 to 1.62 mSv per year. Professional duties of the surveyed employees include the need to make and implement responsible decisions.

Practical experience shows that out of the total number of employees who have passed the psychophysiological examination, 25.1% have a significant reduction in psychophysiological adaptation, which is the risk of developing emergency situations due to the human factor and indications for rehabilitation measures. After completion of the rehabilitation course, repeated examinations were conducted, according to which 92.3 % of employees have a positive dynamics of the level of psychophysiological adaptation, which allows them to be admitted to work.

Thus, the mandatory psychophysiological examination of the personnel of the nuclear industry is an effective element of the radiation protection system, which allows to influence the risks of emergency situations caused by the human factor and to carry out targeted rehabilitation measures in a timely manner.

Keywords: psychophysiological examination, radiation protection, nuclear industry

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OS3.2 (T3.1-0065)

Radioprotection measures adopted during the transportation of spent fuels from the DCMFEI to FACIRI and collective and individual doses evolution analyses

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The purpose of this work is to describe the different stages developed during the transportation of spent nuclear fuel elements (EECC), from the Central Fissionable Special Material Storage Facility (DCMFEI) to the Research Reactor Irradiated Fuel Storage Facility (FACIRI), emphasizing on radioprotection. The results of individual and collective doses will be analysed as well.

The National Programme for Radioactive Waste Management (PNGRR), which is part of the National Atomic Energy Commission (CNEA), is responsible for the safe management of radioactive wastes and spent nuclear fuel elements, which come from nuclear activities carried out in Argentina. They are stored and treated in the Ezeiza Radioactive Waste Management Area (AGE).

AGE is located at the Ezeiza Atomic Center (CAE) and its venue is eight hectares long. Different facilities operate in the venue, one of them is the DCMFEI. Spent nuclear fuel elements generated during the RA-3 Argentinian Nuclear Research and Radioisotopes Production Reactor operation are stored in this facility, adding up to 185 units in July 2015.

In order to increase the capacity and improve the storing conditions of EECC, all necessary tasks were performed to transfer the 185 EECC from the DCMFEI to FACIRI, located at CAE. The main goal all the times was to minimize the doses that workers could receive during a normal operation or non-normal situations, to limit the incident probabilities, to prevent possible accidents that could occur, and, in case they would occur, to establish the actions that should be adopted to mitigate their consequences.

Throughout its development, the operation involved different systems which tend to optimize the radioprotection.

OS3.2 (T3.1-0349)

Qualification of Radiation Workers in KSA, Historical Prospective

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Qualification of radiation workers in industrial, medical, research and services started in 2002. At that time, radiation safety officer (RSO) examination and authorization began. In 2016, a new qualification requirements and procedures were developed to examine RSO candidates, and to issue certificates to individuals who have satisfied these requirements. In addition, certification of qualified experts (QE) was established and up to now about 20 qualified experts were recognized in many radiation practices based on detailed requirements including experience in the field. On the other hand, in 2019, authorization of workers in radiation practices (medical, industrial, research and services) were also initiated. In this paper, lessons learned from implementation of requirements for authorization of RSO, QE and workers will be presented.

OS3.3 (T3.3-0344)

Optimization of Radiation Protection by the Implementation of Local Dose Constraints at ANSTO

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The IAEA tells us that a dose constraint is a boundary value for the optimization of planned individual exposures from a single, specific source or practice, above which it is unlikely that protection is optimized. In the workplace, a dose constraint (better described as a dose review level) is an operational tool used in the optimization of protection.

The Australian Nuclear Science and Technology Organisation (ANSTO) is the centre of Australia's capabilities and expertise in nuclear science and technology. Through its nature an organisation like ANSTO encompasses the wealth of the health physics field and utilises a variety of radiation sources. These include operating the nation's only multi-purpose reactor, OPAL, in Sydney and the Australian Synchrotron in Melbourne; large fixed gamma sources; x-ray machines; particle accelerators; neutron sources; neutron guides; unsealed radioisotopes used in biomedical and chemical applications, biomedical tracer studies, and production settings; and naturally occurring radioactive material (NORMs). As a result, planned exposures are not uniform across or within these practices.

Previously at ANSTO, to ensure that source-related restrictions provide sufficient protection where there are multiple sources, the effective dose to any occupationally exposed person as a consequence of planned exposure situations was constrained to less than 15mSv per year. A review of occupational exposures showed that it is no longer appropriate to have one dose constraint. ANSTO now develops, in consultation with the practice owners and Radiation Protection Experts, constraints which represent the prevailing circumstances. Once determined and agreed upon these constraints are not static and shall be reviewed annually, taking the planned operations for the coming year into account.

This paper discusses the development of source or practice specific dose constraints for optimization and reviews lessons learnt.

Keywords: *constraints, optimization, exposures*

OS3.3 (T3.2-0559)

KNOWLEDGE AND PRACTICE OF JUSTIFICATION OF MEDICAL EXPOSURES AMONG MEDICAL AND DENTAL PRACTITIONERS IN NIGERIA

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Introduction: Radiation protection in medicine is supported by the concepts of justification, optimization and dose limitation. One of the basic principles of radiation safety is to make sure that all exposures to ionizing radiation are clinically justified. Justification is the process of weighing the potential benefit of the exposure against its potential detriment to an individual and it is based on evidence of knowledge of hazards associated with exposure and the clinical information of the patient.

Materials and Method: The study is a prospective cross sectional study conducted among medical and dental practitioners in Aminu Kano Teaching Hospital and Abdullahi Wase specialist hospital within Kano metropolis, Nigeria. A convenient sampling technique was adopted. Data was collected by the use of a semi structured questionnaire, which were administered to them and later retrieved after it was filled. Descriptive statistics in form of mean and percentages were obtained while Pearson's correlation was used to establish relationship between knowledge and practice. Statistical significance was set at $p<0.05$.

Result: There was response rate of 100%. Distributions of the respondents based on specialization were109 (60.6%) medical practitioners and 71(39.4%) dental practitioners. Based on gender, there were 122(67.8%) males and 58(32.2%) females. On assessment of knowledge of justification of medical exposure, majority of the respondents have adequate knowledge with 84.4% medical practitioners and 64.6% dental practitioners. Practice of justification of medical exposure was found to be very poor, with 10.1% medical practitioners and 25.4% dental practitioners. There was a weak positive correlation ($r=0.144$) for medical doctors while a moderate positive correlation was also found for dental practitioners ($r=0.403$). It was found that knowledge was inadequate with medical practitioners having 19.3% and dental practitioners having 42.3% as the only ones aware of it.

Conclusion: knowledge of medical exposure is adequate among medical and dental practitioners while practice of justification is very poor. Knowledge of radiation safety and hazards was also found to be inadequate among them.

Keywords: Ionizing Radiation, Radiation Protection, Justification, Optimization, Dose limitation, Medical exposure

OS3.3 (T3.3-0632)

Quality assurance as a tool for optimization of radiation protection in diagnostic radiology in two tertiary hospitals in low-middle income country

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Background: Quality assurance in diagnostic radiology and its effects in the optimization of radiation protection are well known, but in practice, especially, in low- and middle-income country, there seems to be little or no effort made towards optimization of radiation protection in diagnostic radiology, through quality assurance.

Objectives: This study evaluated the parameters of quality assurance in diagnostic radiology in line with Bonn-call-for-action 2012. The objectives were to evaluate: radiology equipment procurement process; quality control checks and measurements; competence of equipment operators; efficiency of information and communication systems; shielding design goal; and film reject analysis.

Methodology: A cross sectional study design was conducted in two Radiology Centres. Data on equipment procurement were collected using WHO recommended checklist and a questionnaire. quality control checks through visual inspection of the x-ray equipment and quality control measurements were collected through: Kilo Voltage peak (kVp) accuracy and reproducibility test using kV meter (Gammex, model RMI 245, USA); timer accuracy and reproducibility test using Digital x-ray timer (Gammex 07-453); light-beam alignment test using radio-opaque markers and data capture sheet. Competence of equipment operators and efficiency of information and communication systems were assessed. Data on shielding goal design were collected through a Survey metre (RadEye G-10) and data on film reject were collected using adapted WHO data capture sheet. The pointer to optimization of radiation protection in this study was film reject rate of less than 5%. Data were analyzed using summary statistics such as percentages, mean and standard deviation with the aid of a statistical software, Statistical Package for Social sciences (SPSS) version 20.

Results

Parameters of quality assurance evaluated in the study

Centre	Equipment procurement process	Visual inspection of installed equipment (1.6 – 2.0)	Quality Control test: kVp accuracy (±5%)	kVp reproducibility (±5%)	Timer accuracy (±5%)	Timer reproducibility (±5%)	Light beam test (1%)	Competence	Shielding goal survey (0.1 and 0.02 mSv/wk)	Reject rate (<5%)
A	NA	A	NA	A	NA	A	NA	A	A	NA
B	NA	A	NA	A	A	A	NA	A	A	NA

Key: A = Acceptable, NA = Not acceptable

Conclusion: Findings from this study shows that quality assurance practices were not in accordance with the Bonn-call-for-action 2012. Consequently, the optimization of radiation protection in the centres studied were inadequate.

OS3.3 (T3.2-0534)

Using the Alpha-value to Check Overspending

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In this contribution overspending in radiation protection is discussed. Three practical cases are elaborated using the concept of the alpha-value.
The contribution starts with a discussion on the alpha-value. Around 1984, as can be read in the ICRP 42 Publication, it was advised to use this parameter in optimisation calculations. After this publication, the use of the alpha-value became less common, as to obsolete in the past years.
The case is made that the alpha-value can still be a valuable tool to check overspending.
Using the alpha-value, three practical examples are elaborated. They concern the ventilation of a radionuclide laboratory, the use of personal dosimeters in a radiology department and the yearly examination of category A workers. It is shown that in these situations there is serious overspending.
The contributions ends with a discussion on the constraints and possibilities of using the alpha-value. The case is made that the alpha value should be used to check the reasonableness of existing measures in situations where workers receive a dose that is already below -or far below- the dose that is equivalent to a risk in a non-risky occupation.

OS3.4 (T3.6-0591)

Radiological Protection of the Public and Environment: recent developments in the IAEA's Programme

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The paper presents an overview of recent IAEA activities on development and application of international safety standards on the radiological protection of the public and environment.

The development and application of international safety standards on radiation and nuclear safety are important statutory functions of the IAEA. The international standards on radiation safety have a broad scientific and practical basis, such as the scientific findings of the United Nations Scientific Committee on the Effects of Atomic Radiation (UNSCEAR), principles of the radiological protection developed by the International Commission on Radiological Protection (ICRP), extensive international and national experience in maintaining radiation safety.

In accordance with its statutory functions and recommendations of its Member States, the IAEA published General Safety Requirements Part 3 NO. GSR Part 3 on Radiation Protection and Safety of Radiation Sources: International Basic Safety Standards in 2014. This Standard was jointly sponsored by the EC, FAO, IAEA, ILO, OECD/NEA, PAHO, UNEP and WHO. The Standard includes requirements on protection of the public and environment in planned, existing and emergency exposure situations. The IAEA also develops Safety Guides that provide recommendations and guidance on how to comply with the safety requirements.

In 2018, the IAEA published three Safety Guides to help Member States ensure that radioactive releases to the environment from facilities discharging radionuclides are assessed, controlled and, where necessary, authorized, to protect the environment and to ensure that the public who may be exposed, is adequately protected. This IAEA guidance helps regulatory bodies in three ways: to create a framework for protection of the public and the environment [1]; to control and authorize radioactive discharges [2]; and to conduct an environmental impact assessment, which determines whether the planned facility or activity complies with current legislative and regulatory requirements on the protection of the public and the environment [3].

The IAEA engages in activities to promote Safety Guides in Member States and to assess challenges that Member States have in their implementation and the assistance is needed, which is provided in activities such as through on-going training mechanisms and national and regional Technical Co-operation projects co-ordinated by the IAEA. Another key on-going IAEA programme, the most recent of which is called MODARIA (Modelling and Data for Radiological Impact Assessment), assists Member States to build experience in assessing the impact of radioactive releases to the environment. This is achieved by acquiring improved data for model testing and comparison, reaching consensus on modelling approaches and data, development of improved methods and exchange of information.

Keywords: Radiological Protection, Public exposure, Environmental Protection

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OS3.4 (T3.6-0076)

The impact of radionuclides discharged from Koeberg Nuclear Power Station on the marine environment, Western Cape

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Koeberg Nuclear Power station generates about 6% of the total power capacity in SA using Nuclear energy. During normal operations, KNPS discharges liquid and gaseous effluents to the environment comprising various radionuclides which are authorized and regulated by the National Nuclear Regulator (NNR). Historically the system of radiation protection focused on the impact of ionizing radiation on humans with the assumption that non-human species are also protected. The system of radiation protection has over the years changed and there is a need to demonstrate that non-human species are also protected from the effects of ionizing radiation. This is mainly driven by the recent development internationally (ICRP 91, 103,108 and 124).

Therefore, it is of importance to determine the impact of the ionizing radiation on marine organisms living in and around Koeberg Nuclear Power Station. For this study, the ERICA TOOL is used and the model comprises of three Tier levels (i.e Tier 1, 2 and 3). Only Tier 1 and Tier 2 levels were used in this study in-order to assess the impact of radionuclides released from KNPS on the marine environment. The ERICA screening dose rate of 10 µGy/h was applied for all species analysed.

From the result, the model predicts that there is negligible risk to all the organisms considered in this study. Sensitivity analysis of the ERICA results confirms that the polychaete worms are the most vulnerable organisms. The outcome of this study indicated that the highest predicted dose rate is for the polychaete worm with the total dose rate of 1.58 µGy/h from the discharges released by Koeberg Nuclear Power plant during normal operations which is less than the AADQ limits (164.48 µGy/h) and IAEA Benchmark (400 µGy/h).

OS3.4 (T3.6-0182)

Case Studies on the Regulation and Management of Radioactivity in Drinking Water

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The National Nuclear Regulator (NNR) regulates amongst others mines and contaminated legacy sites, associated with Naturally Occurring Radioactive Material (NORM) facilities and activities as part of its mandate. During mining operations and processing of gold in the Witwatersrand basin, uranium and its radiogenic progenies are produced as residual products, which are capable of contaminating the surrounding water resources. The public residing in the vicinity of these areas sites rely on water resources for amongst others drinking water, and this poses a threat to their health and environment. To ensure that the public and environment are protected from radiologically damaging effects, NNR needs to establish regulatory standards and criteria for radioactive levels in the water. Currently, NNR does not have control measures for drinking water in the vicinity of authorised NORM-producing mines and legacy sites. To assist NNR, this research intends to provide a benchmarked case study of international practices in the establishment of regulatory criteria for radioactivity in drinking water. Through this, potential gaps are identified and recommendations made to NNR on effective control measures. The research findings form part of NNR's broader research initiative to assess the natural radioactivity in drinking water in the surrounding vicinities of used and disused gold mines.

Keywords: *radioactivity, drinking water, regulation*

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Oral Session T3

OS3.5 (T2.1-0263)

"NDR: Prototype of National Dose Registry for Latin America. Main experiences in their design and implementation"

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The IAEA Safety Standards require that regulatory authorities take actions for the establishment, conservation, and recovery of the results of radiological surveillance of occupational exposure. Several national regulations and good international practices promote the existence of a central register with results of the individual radiological monitoring. In the majority of the countries (Latin American region), the dosimetry data were neither harmonized, nor managed by a unique database, therefore the possibility of using such data, for proper characterization of the radiological conditions, integrally at national level, in a territory or a sector, and by practices or occupational categories, was very limited. The dosimetric information was not used in the optimization processes of occupational radiological protection.

All these justified the need for designing and developing a prototype of National Dose Registry (NDR) for Latin America, in the frame of IAEA projects RLA9066 and RLA9075. The main objectives of the project were addressing the strengthening of the safety supervision in nuclear applications and the surveillance system for the occupational exposure as well as the centralization of personal dosimetry data, compiling all doses evaluated by the different dosimetric services providers. Experiences of Cuba and Brazil in the implementation of their national registers were well known and accordingly used as bases to design this software. The first version of the NDR prototype, developed in the Spanish language has been designed, developed, and validated. So far, sixteen countries have started to implement the National Dose Registry (NDR).

The present work describes aspects such as the technical bases of the NDR's design, its characteristics, and functionalities. Experiences gained during the implementation in Latin America, problems detected limiting its effective implementation and future challenges are also addressed. Having a common system of dosimetric information management in the region has opened an important space for scientist exchange between the countries and their competent authorities.

The NDR will allow to centralize and preserve the dosimetric history of all occupationally exposed workers of the country (legal character). Its implementation in Latin America has provided regulatory authorities with a tool that permits the verification of the level of compliance with the dose limits and restrictions, as well as to carry out statistics assessments of the results of the individual radiological surveillance that may permit to identify the appropriateness and effectiveness of the radiation protection programs implemented in the practices and to contribute to their optimization

The situation of Latin America has also been observed in other regions (Asia Pacific and Africa). Currently most of the developing countries have not established their national dose registration systems. The first English version of the NDR has been developed based on the achievements and experiences obtained in Latin American region.

Keywords: *National Dose Registry, Dosimetric History, Individual Radiological Monitoring*

OS3.5 (T3.1-0315)

Learning from the Australian Radiation Incident Register

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Australia shares lessons from incidents nationally through its Australian Radiation Incident Register (ARIR). Recent reports from the ARIR highlight lessons from nuclear medicine (2017 report) and the use of computed tomography (CT) scanners (2018 report).

The Australian Radiation Protection and Nuclear Safety Agency (ARPANSA) is the federal regulator for, and Australian Government's primary authority on, radiation protection and nuclear safety. We manage the ARIR to enhance patient outcomes and worker safety. As the sole radiation regulatory body with national reach, we combine the de-identified incidents that are reported via independent jurisdictional regulators. This puts us in a leading position to share information across the nation.

ARPANSA's annual incident summary reports, published on our website and distributed through professional societies, regulatory agencies, industry professionals, and media are a key part of our public engagement on incidents. Incidents are primarily reported from the medical industry, including nuclear medicine production and use, radiotherapy and medical imaging. The presentation will cover a number of recent lessons for sharing, including from recent INES level two and three incidents.

Enhancing worker and patient safety requires getting the right information to the right people. While it is important that radiation incidents are managed on a local level, learnings should be shared across institutions and jurisdictions. To achieve this the register has been undergoing a series of transformational changes to develop an innovative sharing platform across jurisdictions, practices and modalities. Through the register we have an opportunity to connect with diverse institutions, professionals, and government bodies. We have been steadily upgrading the system to make it easier to submit, analyse, and enhance data accessibility. ARPANSA has a track record of successfully delivering high impact projects in the medical space including Australian Clinical Dosimetry Service, Australian National Dose Registry, and the Radiation Protection of the Patient (RPOP) Training Module. ARPANSA has been approached by professional bodies in radiation oncology to, as part of the ARIR, collect near-miss and other oncology event reporting nationally. A key focus is the use of web and cloud based systems and the enhanced involvement of professional bodies.

We are working with oncology groups on integrations directly into hospital reporting systems and other innovative information sharing arrangements. These enhancements will dramatically increase the volume of data through near-miss event reporting, and through increased engagement will allow us to feedback useful information to drive change. As demonstrated by reporting systems such as Safety in Radiation Oncology (SAFRON) and the Radiation Oncology Safety Information System (ROSIS), near miss information and aggregate analysis can lead to valuable information on the types, causes and detection of mistakes, which helps to prevent serious incidents.

ARPANSA has an important role in in the promotion of good radiation protection culture (culture for safety). The incident register collects information on contributing causes to aid understanding in cultural and organisational factors that affect radiation protection performance. We publish a holistic safety guideline and toolkits to help organisations assess their performance and have recently published an assessment of our own regulatory culture.

Keywords: Incidents, Medical, Reporting, Culture

OS3.5 (T2.D-0090)

Radiation Protection Culture in Canada with Seven Decades of National Dose Registry

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Canada has a long history in occupational radiation protection, as evidenced by seven decades of National Dose Registry (NDR) operations. NDR contains the dose records of individuals monitored for occupational exposures to ionizing radiation dating back to the 1940's. The Radiation Protection Regulations [1] require licensees to control doses to workers and to the public and to ascertain these doses. Dosimetry service licensees are obligated to file information with the NDR regularly. Currently, the NDR has records for over half a million workers, including well over 100,000 currently monitored. The NDR mandate is to assist in regulatory control by notifying regulatory authorities of overexposures within their jurisdiction; to evaluate dose trends and statistics to answer requests from regulators and others; to contribute to health research and to the scientific knowledge on risks from occupational exposure to ionizing radiation; and to provide dose histories to individual workers and organizations for work planning and for compensation and litigation cases.

A good example of effective radiation protection measures at workplaces is the significantly decreased radon levels in uranium mines. For uranium miners, the annual average radon exposure decreased from about 1 WLM in 1980's to about 0.1 WLM in recent decades. Miners exposed to high levels of radon underground are a thing of the past. Current miners are exposed to similar levels of radon in their workplaces and in their homes. Nowadays, we recognize that radon protection is not limited to workplaces; rather, it is required for all indoor environments, including residences. Therefore, the need for a radon registry as an add-on to the NDR was proposed. In order to better assist in research and risk assessment, it may be desirable to establish a link between the NDR and existing national radon database.

With increased use of advanced radiation technologies in medicine and a remarkable increase in radiation doses to patients in recent decades, radiation exposure among medical professionals becomes a new focus in occupational radiation protection. A recent study [2] showed that, averaged over 14 years (1993-2006), the annual positive dose received by nuclear medicine technologists was around 2 mSv while the values were below 1 mSv for all other medical professions. Updated analysis on trends of radiation exposure among six medical job categories will be reported here. The results will provide further guidance on practical radiation protection in medical field.

In addition to the two job categories highlighted above (uranium mining and medicine), we will provide the most recent twenty-year trends of mean annual effective doses for all job sectors in the NDR, in order to better understand the effectiveness of radiation protection in workplaces, and to identify places where improvement may be required. The job sectors in the current NDR are accelerator, industry, medical, mining, nuclear and others.

Further, we will discuss new initiatives for NDR to enhance its function of serving the research needs. This also includes updating job categories from two-levels to three-levels in order to better align with the job classes in the UNSCEAR Global Survey of Radiation Exposure.

***Keywords**: Radiation protection, radiation dose, occupational radon exposure*

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OS3.6 (T3.7-0057)

Dose Limits for Occupational Exposure to Ionising Radiation and Genotoxic Carcinogens – A German Perspective

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This paper summarizes the view of the German Commission on Radiological Protection ("Strahlenschutzkommission", SSK) on the rationale for limits of effective dose from occupational exposures to ionizing radiation.[1] It includes a discussion of the reasoning behind current dose limits followed by a review of studies on radiation-induced cancer and cardiovascular disease. To put these concepts in perspective, actual occupational radiation exposures are exemplified with data from the dose registry in Germany. Procedures currently in use in Germany, Switzerland and the USA to define limits for occupational exposures to genotoxic carcinogens are discussed. For the regulation in Germany it is noted that the lifetime effective dose limit of 400 mSv corresponds to a cancer risk of 0.04, one order of magnitude larger than the risk tolerated for occupational exposures to single genotoxic carcinogens. The SSK recommends i) continuing efforts to standardise the terms and concepts used to derive dose limits in a range of different workplaces; ii) keeping a lifetime effective dose limit in Germany but discuss its numerical value; iii) initiating an international discussion of the limit of 100 mSv effective dose in five years, because risks considered tolerable by the society have changed over the past decades; and iv) keeping the annual dose limit of 20 mSv effective dose.

***Keywords**: Occupational exposure, dose limit, cancer risk*

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Oral Session T3

OS3.6 (T3.7-0132)

Can we compare the excess risk due to carcinogenic substances and the detriment due to ionizing radiation exposure?

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More and more radiation protection experts attempt to compare chemical risks with radiological risks to promote a graded or commensurate approach of the radiological risk with respect to other risks in the framework of impact studies at low exposure. For no-threshold substances, a shortcut can be readily made by comparing the risk figured out by the product of the exposure to a chemical substance by its TRV (toxicological reference value) and the detriment as the product of the radiological dose by the nominal risk coefficient adjusted to the detriment. This presentation aims at explaining how the TRV (slope factor) and the nominal risk coefficient adjusted to the detriment are built. The main differences between the two patterns are outlined (ie the critical effect on an organ for chemical substances versus the whole body for ionizing radiations, occurrence of an adverse effect versus detriment which takes into account the consequences of the disease…). The end of the presentation tables some proposals to allow the comparison between chemical substance impact on health and ionizing radiation impact. These proposals can be slightly different whether a single substance or a mixture of substances are at stake. These proposals are made considering the different steps of the ICRP model to go from cancer cases in the lifespan study to the effective dose.

OS3.6 (T3.7-0164)

ICRP activities related to veterinary applications of ionizing radiation

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After Rontgen's discovery of X-rays, veterinarians were among the first to perceive the potential benefits of radiology for animal health care. For many decades, plain film radiography was about the only application. The number of procedures was limited, and the doses involved were low. Doses to human bystanders would be low to trivial provided some simple rules were followed and consequently, veterinary applications were not a priority for radiation protection.

Since then veterinary procedures making use of ionizing radiation have substantially increased in numbers, and modalities are now as diversified as in human health care. The number of exposures in diagnostic radiology has increased, including as a result of digitalization, and higher dose applications such as CT- or CBCT-scanning are becoming readily available in many centers throughout the world. Interventional radiology procedures have entered the practice field, and so have nuclear medicine applications, both diagnostic and therapeutic. Finally, radiation oncology is available in an increasing number of veterinary clinics around the world.

As a consequence of these important practice changes, the radiation-related risks have also increased and diversified. For instance, the risk of contamination by radioactive substances to staff, owners, handlers and to the environment from nuclear medicine procedures will now require our attention. Radiation doses to veterinary practitioners performing interventional procedures need to be closely monitored since they could be far from negligible, as could the doses to the animal patients themselves, based on experience with similar procedures on humans.

There are many unique radiation protection challenges in veterinary practice compared to human medicine. For example, many radiological procedures on large animals are performed outside of a regulated environment and new equipment may have been retrofitted in an existing building without due consideration of shielding requirements. Justification is not supported by a veterinary equivalent of the "referral guidelines" or "appropriateness criteria" we know from human medicine, there are no DRL's , there is little to no agreement on activities of radiopharmaceuticals to be administered for therapy purposes, there is no involvement of a medical physicist, and last but not least, not all practitioners performing higher dose diagnostic or even radiotherapy procedures have had specialized education and training.

Task Group 107 had the first charge from the ICRP, tasked with deciding if veterinary practice warranted specific radiation protection considerations. This task group made the above observations and recommended that the ICRP should indeed devote attention to this practice area. Consequently, the ICRP created another task group (TG 110) and charged it with developing an initial set of recommendations for radiation protection in veterinary practice. The mandate of TG 110 is expansive; its primary field of attention will of course be the protection of humans involved in the procedures, both professionals and members of the public, but the animal patient's protection will also be considered. Additionally, TG 110 will also consider contamination of the environment as a consequence of nuclear medicine applications.

Representatives from TG 110 will present its draft recommendations to the IRPA 15 Congress and request feedback from individual radiation protection professionals and from IRPA's Associate Societies.

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Oral Session T3

OS3.6 (T3.1-0512)

An Urgent Need for the Simplification of the Present System of Radiation Protection Quantities in Applications of Radiation and Nuclear Technologies

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During its more than one hundred long history, the system of the quantification of radiation exposure underwent through a number of phases, where many new elaborated quantities reflecting understanding and the level of knowledge of radiation effects at that time were applied. As it happened, newly introduced quantities, which were usually defined in an even more complex manner than previous ones, were always for some time used together with the older quantities. Moreover, in the definition of novel quantities emphasis was paid much to their too strict relation in reflecting biological effects and less attention to the problems of their direct measurement and monitoring. This is why we have now in use quantities recommended for the regulatory control of radiation exposure, which cannot be readily experimentally and reliably assessed in practice. Moreover, there are too many quantities currently used in radiation protection where there is also some confusion in relevant units where quite often the unit Sv is applied often to quantify high exposures characterized by deterministic rather than stochastic effects. The paper presents an overview of the existing situation in this field and proposes some changes. It is suggested to limit the use the current, rather complicated system of radiation protection quantities by scientists at universities and research institutes, while for routine monitoring to develop and adopt a much simpler system based on measurable quantities. Anyway, one has to admit that the present system of quantities is so intricate that even radiation workers engaged in routine use of radiation and nuclear technologies cannot fully appreciate and implement it, including correct interpretation of the results of their measurement and monitoring.

OS3.6 (T3.7-0353)

Individual Radiation Protection: Idea and Needs for Research

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The current Radiation Protection System is based on the definition and use of the "effective dose", a complex quantity which is not measurable and normally not easily understood by the general public.

National radiation protection requirements and regulations stem from a few basic principles, one of which is the respect of individual annual dose limits, specified in terms of the quantities "effective dose" and "equivalent dose". Another guiding principle is the continuous quest for optimization of individual integrated doses, i.e. trying to keep radiation exposures "low", assuming that any exposure to radiation is an increase of radiological risk to the exposed individual.

Despite the general robustness of the Radiation Protection System, which derives from more than a century of research and experience in the various uses of radiation, some of the fundamental assumptions upon which the Radiation Protection System is built also constitute the source of its present weaknesses.

The paper discusses four such weaknesses (the disconnect between "design" doses and "actual" doses; the use of the quantity "effective dose" as a proxy for radiological risks; the use of the "Linear, No Threshold" hypothesis and of the "optimization" principle; the use of the "reference person" concept), and proposes research that could lead to the development of a new Individual Radiation Protection System framework that (i) implements a scientifically sound correlation between exposure to ionizing radiation and health risks, (ii) leverages recent advances in computational modelling and genomic sciences, and (iii) has the potential to lessen the costs and regulatory burden for radiation workers in all fields.

Keywords: *individual RP, effective dose*

OS3.7 (T3.1-0543)

The Inter-Agency Committee on Radiation Safety – 30 years of international coordination of radiation protection and safety

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The Inter-Agency Committee on Radiation Safety (IACRS)[1] was constituted in 1990, as a forum for collaboration and coordination between international bodies with regards to radiation safety. It consists today of representatives of eight intergovernmental member organizations (EC, FAO, IAEA, ILO, NEA, PAHO, UNSCEAR, WHO) and five observer non-governmental organizations (ICRP, ICRU, IEC; IRPA, ISO). The IACRS provides a platform for interaction between these relevant international bodies to contribute to a common understanding of the scientific basis and legal framework for the application of the system of radiological protection, towards global harmonization of radiation safety standards. The IACRS played a key role in the development of the International Basic Safety Standards (BSS) in 1996 and in its recent revision in 2014.[2,3] Further, an IACRS specific Task Group—chaired by the IAEA—fosters the implementation of the BSS in a consistent and coherent manner in all Member States of the United Nations.

The IACRS operates via a standing secretariat jointly provided by the IAEA and OECD/NEA and is chaired by one of its member organizations on a rotating basis for periods of about 18 months. This approach has proved to be effective and was the foundation for ensuring continuity of the work of the committee and at the same time allowing a rotating leadership for all member organizations. Currently, the IACRS is chaired by WHO.

The International Radiation Safety Framework under which the IACRS works is structured around four main areas: (a) Science; (b) Principles; (c) Standards; and (d) Practice. This paper presents briefly the mandates, roles and functions of the various international bodies that are relevant to the four above mentioned areas of work, discusses how these bodies coordinate their actions and complement each other to enhance radiation protection and safety worldwide and describes their contribution to the achievement of the Sustainable Development Goals (SDGs)[4]. The paper also provides an overview of the main accomplishments of the IACRS since its inception 30 years ago, and an outlook on key challenges for its future activities.

Keywords: *International Basic Safety Standards, Radiation Safety, Sustainable Development Goals*

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OS3.7 (T3.7-0165)

Review and Revision of the System of Radiological Protection

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The International Commission on Radiological Protection (ICRP) has begun a review of the System of Radiological Protection working towards an update of the general recommendations. Initiated in 2018 with a brainstorming session within the Main Commission, this entails a comprehensive review of all elements of the system, including the aims, scope, principles, key concepts, tools and requisites, and the scientific and ethical bases. This is very likely to lead to a revision of the current system presented in ICRP *Publication 103* (1).

The review and revision of the system that resulted in ICRP *Publication 103* took a decade, supported by ICRP's first comprehensive, open consultation process. The review and revision of the current system is likely to take at least as long, and there is no doubt that engagement with the radiological protection community will be even more extensive. ICRP is currently developing some of the key questions to be addressed during the comprehensive review, raising awareness that this review is beginning, and considering how best to manage its programme of work and the broad engagement with organisations and individuals essential to the process.

Some of the building blocks for a revised system are already under development, for example through ICRP Task Group 91 Radiation Risk Inference at Low-dose and Low-dose Rate Exposure for Radiological Protection Purposes, ICRP Task Group 111 Factors Governing the Individual Response of Humans to Ionising Radiation, and ICRP Task Group 114 Reasonableness and Tolerability in the System of Radiological Protection. Workshops and consultations associated with these Task Groups are an essential part of engagement on the review and revision of the system. We anticipate addressing further building blocks through additional Task Groups in the coming years, each with its own mechanisms for engagement. The biennial ICRP symposia on the system of radiological protection will also be important venues for this, especially later in the process as drafting of the new fundamental recommendations begins.

It is difficult to predict what might be revised in advance of the review. However, at present, it looks like the revision might focus on: consolidating developments since the 2007 recommendations e.g. protection of the environment, post-accident recovery, and a decade of experience with planned, existing, and emergency exposure situations; explicitly referencing the ethical basis of the system elucidated in ICRP *Publication 138* (2); reflecting the latest scientific understanding of radiation effects, especially at low doses and dose rates and considering factors that influence individual response; and, clarifying the system overall. In addition, it will be necessary to consider the implications of emerging domains needing radiological protection guidance, e.g. veterinary patients and space travel.

Keywords: radiological protection, general recommendations

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OS3.7 (T3.B-0214)

Bo Lindell's History of Radiation, Radioactivity, and Radiological Protection

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Professor Bo Lindell (former UNSCEAR Scientific Secretary, inaugural laureate of the IRPA Sievert Award, and, perhaps his best remembered distinction, former Scientific Secretary and then Chair of ICRP) retired from his position as head of the Swedish Radiation Protection Authority in 1982 but remained actively involved in radiological protection until shortly before his passing in 2016. During his 'retirement', he researched and wrote his *magnum opus*, a history in four volumes of radiation, radioactivity, and radiological protection, comprising over 2 600 printed pages published in Swedish from 1996 to 2011. The style is witty, entertaining, and spiced with many anecdotes about the personalities involved. Lindell's description of the development of nuclear weapons (described by a reviewer as 'exciting as a thrilling detective story') is a unique and well-thought-out compilation from many sources; the other parts are based largely on first-hand participation.

Lindell was extremely well placed to write this history. He met many of the pioneers in the field such as Lauriston Taylor, Gioacchino Failla, and Hermann Holthusen, and became one of the leaders of the next generation devising the current System of Radiological Protection together with his best friend Dan Beninson, John Dunster, and David Sowby. This personal experience brings his stories and the characters in them to life, making the books a fascinating and essential read for anyone interested in the history of science, expert or layman alike, especially those interested in radiological protection.

Aided by generous grants from IRPA and other sources, the Nordic Society for Radiation Protection (NSFS) has organised a complete translation into English of this authoritative work. The first part, *Pandora's Box*, covers observations and perceptions of radiation all the way from ancient Greece, through Röntgen's 1895 discovery, until the demonstration of nuclear fission in the late 1930s. The second book, *The Sword of Damocles*, deals with developments during the 1940s, dominated by the enormous efforts spent on atomic bomb research, but this was also the time when much of the current philosophy in radiological protection was founded. *The Labours of Hercules*, the third tome, treats the period from 1950 to 1966 when the intensity of Rolf Sievert's Herculean efforts peaked, a time of huge scientific progress but also of awakening in the shadow of 'doomsday bombs' and global radioactive fallout. The final volume, *The Toil of Sisyphus*, on the time from 1967 until 2008, is characterised by Lindell's own participation as one of the world leaders in radiological protection as well as by the endless efforts of the emerging nuclear power industry to gain public acceptance.

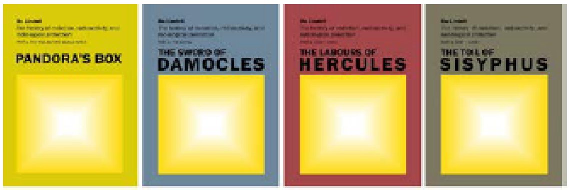

Fig. 1. Printed versions are available through Amazon for the price of printing and shipping only, and PDF versions can be downloaded for free from www.nsfs.org and www.nks.org.

Keywords: Information, Education, Philosophy

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Reflections on Low-dose Radiation – Misconceptions, Reality and Moving Forward

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Radiation is a constant variable in our daily lives that most people do not usually consider when going about their everyday activities. Recent documentaries on 1986 Chernobyl or the 2011 Fukushima accidents have brought the topic of radiation to the forefront of conversation in the public square once again. Even so, there currently exists an undeserved perception gap surrounding low doses of radiation linked to an increased risk to public health. Despite the dramatic portrayals on the television screen, operating nuclear power plants demonstrate a remarkable safety record of normal day-to-day operations. They operate under strict radiation protection standards that require the use of advance technological systems to protect individuals and the environment from the harmful effects of ionizing radiation. Where operating nuclear power plants are in operation, societies exist and thrive experiencing no detrimental effects to human health. In fact, UNSCEAR in 2012 confirmed that an increase of risk to public health cannot be "attributed reliably to chronic exposure to radiation at levels that are typical of the global average background levels of radiation" [1]. A level that exceeds by far exposures from nuclear installations.

Radiation protection standards should be based on the best current scientific understanding of the effects on radiation, as well as proportionate public policy generated through political debate driven by informed public opinion. Over-conservatism applied to further reduce low-dose levels of radiation can be attributed to the misuse of the Linear No-Threshold (LNT) hypothesis and the As Low As Reasonable Achievable (ALARA) principle within the industry. The LNT hypothesize that radiation risk is proportional to dose, so that even doses typical of a person's average exposure (i.e. a few mSv per year) is assumed to carry a small but defined risk of cancer. ALARA is used as an effective method for codifying the LNT dose-reduction model for day-to-day radiation exposure implementation at the reactor site. While ALARA's initial application considered both the socio-economic and public health aspects, current ALARA philosophy and implementation has shifted well beyond its original intent.

Due to the renewed interest among the public, experts and policy-makers on low-dose level exposure to radiation, this timely paper will discuss current knowledge and realities of low-dose radiation as informed by current scientific understanding. Additionally, the paper will explore possible forward trends and future application of radiation protection principles involving low-dose radiation.

Keywords: *Linear no-threshold, low-dose radiation, standards*

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OS3.7 (T3.1-0135)

Radiation protection today: success, problems, recommendations for the future

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In recent years, discomfort on the current situation of radiation protection as well as how it is perceived by society became evident in the radiation protection community including some publications by well-known experts (Gonzales et al. 2013, Coates 2013, Coates and Czarwinski 2017, 2018). These concerns apply to technical aspects of the regulations and their implementation and, to a particular extent, also the perception of radioactivity, radiation, and radiation protection by today's society. It appears necessary to improve information and explanations in the following areas: risk factors for radiation-induced health effects; goals, potential and limitations of epidemiological studies; quantities and units of radiation protection; risks due to internal exposure in comparison to that of external exposure, etc.

The German-Swiss Association for Radiation Protection (Fachverband für Strahlenschutz) has therefore commissioned a working group to comment on fundamental issues of radiation protection and to make recommendations for its future development. The results of the work of this group became a position paper of the FS and are presented here. The paper, consisting of four chapters, was presented earlier to the members of the German-Swiss Radiation Protection Association (FS) for comments. After taking into account the – mostly very positive – comments of the FS members, the work was submitted to the Executive Board of the FS for acceptance as a position of the FS. The PDF can be downloaded from the FS website: https://www.fs-ev.org/themen/. In October 2018, the paper was presented to the European Associate Societies of IRPA at Paris for discussion and comments and, in February 2019, it was discussed during a Workshop of Dutch Society for Radiation Protection (NVS) at Utrecht.

The position paper looks back to the history of radiation protection: the way to more safety. It deals in detail with the natural radiation and its risks as a reference for radiation protection and proposes a lower limit of optimization. Finally, it summarizes the discomfort in the radiation protection community and makes 24 recommendations for the following partially overlapping topics: radiation and radiation risk; radiation doses, radiation effects, and dose limits; practice of radiation protection; dealing with emergency situations and communication with and information of the population.

Keywords: radiation protection, discomfort of the practitioners, recommendations for the future

ACKNOWLEDGMENTS
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OS3.7 (T3.1-0597)

Protection against ionizing radiation *vis-à-vis* non-ionizing radiation: Unjustifiable different approaches; a potential derived ethical conflict

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Aim: The paper objective is to analyze the international scientific bases, the globally accepted protection paradigms and the intergovernmental standards for protection against exposure to both, ionizing radiation (IR) and non-ionizing radiation (NIR). They are compared and conclusions about their differences are drawn. The ethical consequences arising from differences are explored.

Background: The ***international scientific basis for the protection against IR*** is derived from studies by the prestigious United Nations Scientific Committee on the Effects of Atomic Radiation (UNSCEAR), which is providing yearly estimates from 1955 to the highest international body: the General Assembly of the United Nations (UN). UNSCEAR remits does not include NIR and there is not an equivalent international governmental body to provide a thorough international consensus on the global levels and effects of exposure to NIR.

The ***global protection paradigms for protection against IR*** are given by the International Commission on Radiological Protection (ICRP) and the International Commission on Radiation Units and Measurements. (ICRU). The International Commission on Non-ionizing Radiation Protection (ICNRP) was created with similar objectives for NIR although the ICNRP protection paradigm seems to be wholly different.

These differences have created a feeling of '**double protection standards**' between ionizing and non-ionizing radiation. Radiation protection specialists working in IR and NIR, who are associated through the International Radiation Protection Association (IRPA), may face some ethical problems in applying such double protection standards, which could place them in direct conflict with IRPA code of ethics.

Different Scientific Approaches: In the case of IR, the main information on which UNSCEAR estimates are based is provided by the large radio-epidemiological study of the Hiroshima and Nagasaki Cohort (Life Span Study or LSS) Conversely, in the case of NIR, there is not any wide epidemiological study available similar to the LSS cohort, in spite of plenty of evidence the detrimental effects attributable to exposure to NIR.

Anyway, at the current state of knowledge there are no much differences between the biological effects of IR and the NIR, in both cases, DNA damage occurs, which may be the beginning or the first stage of a cancerogénesis process.

It is not understandable that similar risks are controlled so differently. This should at least be justified and explained.

Different paradigms:. The protection paradigms for IR and NIR are quite different. The protection against IR follows a LNT model, which provide protection at much lower doses of those for which there is absolutely no attributability of effects but only subjective inferences of risk. There is not a similar approach for NIR. The main ICRP paradigm for IR protection is optimization of protection; however, ICNIRP feels that optimization cannot be performed for NIR because the cost of health detriment is unknown and the cost of protection is country dependent. The ICRP paradigm is based on the UNESCO precautionary principle; no precautionary approach is used for NIR. Individual doses are used as constraints for individual sources of RI and as limits for the summation of all controllable sources of RI; ICNIRP feels that the difference between limits and constraints tends to create confusion and mistrust of authorities and increase public concern rather than reducing worries and controversies.

Different standards: A safety regime that include a vast corpus of intergovernmental international protection standards exist for IR. There is no a safety regime of international intergovernmental standards for NIR.

Conclusions and Proposals: The protection approaches between IR and NIR are wholly different: different scientific basis; different paradigms; different standards. Radiation protection practitioners might be violating the IRPA Code of Ethics by accepting these differences. It would be convenient to agree on an international consensus on the supporting science needed to determine the risks of IR and NIR and UNSCEAR might extend its remit with this goal. It would be moreover convenient that the same paradigm be applied for protection against physical agents that produces similar or at least comparable biological mechanisms and health detriments. (The criteria applied by the ICRP and the ICNIRP are diametrically opposed both in the way of establishing the cause / effect relationship and in the way of applying the principles of radiological protection, which generates a source of conflict at the time of making decisions.). Furthermore, it would be convenient to have a similar international regime of intergovernmental standards for protection against IR and NIR. Finally, IRPA might promote the preparation of protection practitioners in RNI abrest of the biological effects, the protection paradigm and the regulatory aspects, including operational criteria and measurement systems since, at present, many NIR experts seems to have knowledge in measurement systems but incomplete information on the rest of the topics.

Keywords: ICNIRP, Regulation, Conflict

OS3.7 (T3.C-0380)

Radiation Protection Practitioners – Influencing the Future

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The Head of Profession (HoP) for radiation protection in a company takes the lead in the promotion, development and oversight of a range of radiation protection roles. In 2019, the Society for Radiological Protection (SRP) embraced a group for HoP within the UK set up by its members. By the second meeting, over 23 companies (covering nuclear, medical and consultancy) had representation. The Group is a forum for discussion on the national challenge of recruitment and retention in this field. It is understood that the radiological protection practitioner will be in demand – especially in decommissioning, for many years to come, but there is the awareness of a challenge in recruitment across many of the specialisms. It is agreed that communication to the rising generations needs to modernise. Optioneering and influencing strategies that can be adopted to secure the future of the profession will be described.

OS3.8 (T3.8-0083)

Education and Training of RPOs in the Netherlands – beyond EU 2013/59

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In recent years, the Education and Training system for Radiation Protection Officers (RPOs) in the Netherlands has changed significantly because of the implementation of the European Directive 2013/59/Euratom. The training programs are much more application specific than before. In this contribution, we will comment from a training provider's perspective on this change. It seems to be clear that the new system has both clear advantages and disadvantages above the system in place until 2018. A clear unsolved problem in the new system is the mixing of all measurement and control applications into one set of learning outcomes. We report on the outcomes of a project on behalf of the Dutch Authority for Nuclear Security and Radiation Safety to develop more adequate learning outcomes for RPOs responsible for measurement and control applications.

Keywords: *RPOs, Education & Training, Measurement and Control Applications*

OS3.8 (T3.8-0108)

A Virtual Radionuclide Laboratory for Education and Training in Radiation Protection developed in MEET-CINCH

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The MEET-CINCH project, supported within HORIZON 2020, follows two previous projects (CINCH I and II) with the aim to foster Education and Training in the field of Nuclear and Radio Chemistry and to sustain the results obtained in CINCH I and II. One aim of MEET-CINCH was the development of a tailored training event for members of regulators and administrative bodies. In contrast to the international nature of basic science, needs of regulators differ strongly from one country to the other and many technical terms and regulations require use of the respective national language. In MEET CINCH, a three-dimensional environment corresponding to a radionuclide laboratory was developed in order to take these special requirements into account. After the lab was developed, one of the partner involved in MEET-CINCH, the Institute of Radioecology and Radiation Protection, Leibniz University Hannover, conducted a course with 120 members of regulators from all over Germany. Main task of that course was to visit the virtual laboratory in order to answer the question of whether or not a license for the handling of radioactive substances can be granted for that virtual lab in accordance with German legal requirements. Afterwards the participants evaluated the course. The results of the evaluation were very positive; especially the members of the regulators and administrative bodies appreciated that (a) the usage of the virtual lab is very motivating, (b) the VR-radionuclide-lab is a perfect tool to train especially new members of the regulatory bodies and prepare them concerning important radiation protection issues before entering a radionuclide lab and that (c) the VR-lab can be entered at any time and training sessions in that lab can be repeated whenever it seems to be useful.

In this contribution, the virtual radionuclide lab developed in MEET-CINCH will be presented and the usage in radiation protection courses (including the one designed especially for members of the authorities) will be described. Additionally an outlook concerning further developments and applications (especially the usage of VR-glasses, features like moving objects and measuring dose-rates or contamination in the lab) will be presented.

Fig. 1. The virtual radionuclide lab developed in MEET-CINCH

Keywords: Education and Training in Radiation Protection, Virtual Reality, Radionuclide Laboratory

OS3.8 (T3.C-0528)

The NEA International Radiological Protection School (IRPS)

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While guidance and standards documents describe the technical facts in relation to the radiological protection (RP) system, the body of understanding that they reflect, including how the different elements have evolved, are not well documented. To appropriately and effectively apply the RP system to existing and emerging situations, the "spirit" of the RP system – its nuances and history – needs to be fully understood by tomorrow's RP leaders.

In an effort to respond to this challenge, the OECD Nuclear Energy Agency established the International Radiological Protection School (IRPS), to provide a clear understanding of the RP system and how it is intended to be interpreted for application in diverse and emerging circumstances. Closely with the ICRP scientific secretariat, this initiative has been implemented during summer 2018 and 2019, and will be renewed in 2020. An introductive materials, the experts who contributed to the RP system's creation provide an historical overview of how and why the RP system evolved, as well as a deep understanding of what the system is intended to mean. The programme, held over five days, includes then sessions built on a mix of presentations and illustrative case studies. The programme is aimed at mid-career experts in the field of radiological protection. Participants should hold positions providing policy and practical level advice in government ministries, regulatory authorities, research institutions, nuclear fuel cycle industries or in other industrial or medical sectors where RP expertise is needed. Sessions are attended by *ca*. 40 participants, and are hosted at Stockholm University at the end of August. This presentation will describe the IRPS programme and list its lecturers.

Keywords: *Radiological Protection system, International School, mid-career experts*

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Science, Technology, Society approach to Education of Biologic Effect of Low Level Radiation in Medical School

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Aim/Introduction: Education of biological effects to low level radiation (LLR) which is essential for radiation protection is limited in the curricula of medical schools. Interactive discussion sessions using science, technology, society (STS) approach is designed to raise understanding of LLR in medical school students.
Materials and Methods: Episodes and issues of LLR were selected and used for interactive discussions. Korean social issues such as radiation exposure by monazites in furniture, and international issues such as Chernobyl and Fukushima accidents were used. Students were divided into 50 small groups, which were again divided into pros and cons teams of LLR. Surveys on the perception of LLR were done several days before and after the discussion session. Perception of each questions was scaled to 10. Changes of perception were analyzed by paired t-test.
Results: Majority of students eager to learn the biological effects of LLR. Carcinogenesis of LLR is selected as the most important issue. Current social issues in Korea attracts the students more than international issues. Attitudes on the hazards of LLR were significantly changes. Attitude to exposure by medical radiation was also significantly changed.
Conclusion: Education of LLR by STS approach using episodes of current society issues changed LLR perception of medical students significantly. Understanding of medical radiation exposure was also significantly improved.

Keywords: *Science Technology Society, Medical Education, Low Level Radiation*

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Argentine Radiation Protection Young Professionals Network (RedSARJoven). Engaging the New Generation

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The Argentine Radiation Protection Society (SAR) since its creation, in 1966, is working to fulfil the foundational objective of promoting knowledge on radiation protection through training, dissemination of information and knowledge management. This is why SAR carries out the regular dictation of courses recognized by the competent authorities and also special courses.

In recent times, it has become clear that the young generation has a different way of communication and learning. It is a real challenge to adapt the education and training in radiation protection to the modern world, a technological world, where information is extremely wide and where there are many ways to share knowledge and experiences. Thinking on the needs of the Radiation Protection Young Generation, SAR decided to launch the Argentine Radiation Protection Young Professionals Network (RedSARJoven). This network was created with the purpose of promoting the participation and inclusion of young professionals working in radiation protection in all its applications, improving the resources of the human capital and its competences and promoting the culture for safety in the new generation.

RedSARJoven was launched at the IYNC-WIN 2018 Congress (International Youth Nuclear Congress - WIN Global Annual Conference) in March 2018, in Bariloche (Argentina) at the Panel "Youth in Nuclear Energy - A Nuclear Movement", where the perspective of young professionals working in nuclear energy and radiation protection were presented by leading professionals from international organizations. One of the main objectives of RedSARJoven is to create a space for the exchange of ideas and joint work among the young professionals working in radiation protection and also to encourage young professionals to start working in this area, giving them information about the different opportunities. To fulfil this objective, and taking advantage of the extensive experience of elder members of SAR, RedSARJoven is planning to launch a Mentoring Program for knowledge transfer, exchange of experiences and fundamentally to guide young professionals in their radiation protection career development.

During the 2018-2019 period, RedSARJoven has organized different activities for young professionals, as a "Workshop on Radiation Measurement" and a ¨Workshop on Optimization Applied to Medical and Industrial Practices¨. Representatives of the network had participated in national, regional and international events, like in the XI Latin American Regional Congress in Havana – Cuba, with the aim of share the experience in the region and encourage creating more national networks.

RedSARJoven is in line with IRPA's goals that encourage the participation of young professionals, as well as being part of the global youth movement in all areas, from nuclear energy to medical applications, from industry to education, research and communication.

***Keywords**: Communication, Radiation Protection, Young Professionals Network*

OS4.1 (T4.3-0272)

Use of CT in asymptomatic people for individual health assessment (IHA): improving clinical governance and regulatory compliance

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While advanced imaging technology has improved patient care, a substantial fraction of imaging procedures is unnecessary and does not provide sufficient benefit compared to its harms, costs and ethical implications. Today, evidence-based imaging referral guidelines can assist the decision-making process for patients with clinical signs and/or symptoms that raise clinical suspicion of disease. In contrast, evidence about the complex interplay between the benefits, harms, costs and ethical implications of medical imaging in asymptomatic individuals is still mostly lacking. The requirements in the International Radiation Basic Safety Standards/ BSS say that both- the radiological medical practitioner and the referring medical practitioner- shall be involved in the justification of radiological procedures to be performed in asymptomatic individuals outside population screening programs and the individual shall be informed of the expected benefits, risks and limitations (1). Computed tomography (CT) is being increasingly used in asymptomatic people for coronary artery calcium scoring, investigation of coronary artery plaques, early detection of lung and colon cancers and whole-body CT surveys, in what is referred to as individual health assessment (IHA). These CT-IHA practices are not part of organized screening programs, the evidence of the effects of CT-IHA for people and for society is usually weak or even absent, quality assurance is not always in place, they are often performed outside the healthcare pathway and no arrangements are in place for transferring the results into the healthcare system. The process of justification of CT-IHA requires considerations that go beyond the radiation risks, such as public health impact, health financing, ethical issues, overdiagnosis, overtreatment, false positives, false negatives, indeterminate and incidental findings, among others. CT-IHA challenges the principle of justification but at the same time provides the opportunity of building upon existing requirements and legal instruments for establishing a regulatory framework. To address this challenge, WHO convened four international workshops in Munich, Germany (2014); Seoul, South Korea (2016); Quebec, Canada (2017) and Copenhagen, Denmark (2018) (2). Subsequently, guidance to enhance justification and clinical governance of CT-IHA practices was developed, noting that such practices require greater scrutiny in terms of safety and quality, involving health authorities, radiation protection bodies, healthcare professionals and other relevant stakeholders. It is also acknowledged that notable uncertainties are still involved in assessing both benefits and harms of this practice, as well as in the overall experience of persons in receipt of IHA. A robust well-defined framework for good governance of this practice should consider the evaluation of the justification process and appropriateness of the procedure undertaken; protocols ensuring optimization of protection; quality assurance programs; education, training and performance of staff; documentation and process evaluation; robust and confidential mechanisms for integrating the results into an established care pathway; registration systems to gain information on benefits, harms and other outcomes and provision of that information, in a suitable form, to the scanned individuals before they present for scanning, and to professionals in the wider health system. This manuscript summarizes the structure and content of the new WHO document.

Keywords: radiation protection, medical imaging, asymptomatic individuals, screening

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Oral Session T4

OS4.1 (T4.3-0168)

Pediatric interventional radiology and cardiology in Latin America and the Caribbean (OPRIPALC project). An international effort in optimization.

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When interventional radiology (IR) and interventional cardiology (IC) procedures are performed in children radiation doses may be relatively high. For some complex cases, it might result in tissue reactions such as skin injuries if the X-ray systems are not under strict quality control programs, the operational protocols are not properly supervised, and operators are not trained in radiation protection (RP). An additional problem with some of the procedures in pediatric procedures is the re-intervention rate due to the reappearance of the disease. For a given radiation dose, children are generally at more risk of cancer induction than adults. According to UNSCEAR reports, the lifetime cancer risk for children might be a factor of 2 to 3 times higher than the estimates for an average population. The International Basic Safety Standards (BSS) and the Bonn Call for Action pay special attention to pediatric patients and the justification and optimization.

The ICRP has issued new recommendations on Diagnostic Reference Levels (DRLs) including advice for pediatric interventions (1). The new technology in X-ray systems and post-processing of the images should be implemented with the appropriate training (including the RP aspects) and a regular audit of patient doses and image quality. The radiation risk communication is a relevant aspect in pediatric imaging and especially in interventional procedures and it should be integrated in the training programs (2).

Only a few countries in the world have implemented the use of DRLs in pediatric IR and IC. The project "Optimization of Radiation Protection in Pediatric Interventional Radiology and Cardiology in Latin America and the Caribbean" (OPRIPALC) aims: 1) To promote radiation safety culture in pediatric IR (including training actions); 2) To improve radiation safety and quality of care in the participant centers. 3) To define optimization strategies based on a collection of patient doses from a sample of representative hospitals in different Latin American and the Caribbean Countries for setting DRLs. 4) To produce a regional consensus document on these issues.

The first steps of OPRIPALC have been fulfilled: selection of 36 pediatric hospitals from 10 different countries; selection of 3 frequent procedures for IR and 3 for IC; preparation of training material on radiation protection and a common basic quality control protocol for the X-ray systems. The project expects to enhance optimization of protection in pediatric interventional procedures, identifying the main challenges and proposing solutions.

***Keywords**: pediatrics, interventional procedures, DRLs, optimization*

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OS4.1 (T4.3-0240)

Personalized dosimetry of chest CT using patient models

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The aim of this study was to provide a) accurate 3-dimensional (3D) dose distribution and b) normalized organ radiation doses for adult chest CT examinations for primarily exposed organs. Chest CT examinations performed on 110 adult patients were included in this study. Patient CT image data were used to create 110 patient models. These models were used as input in an equipment-specific and patient-specific Monte Carlo software (ImpactMC, CT Imaging GmbH, Erlangen, Germany). ImpactMC is a well validated Monte Carlo code specifically designed for 3D dosimetric evaluation on CT acquired images (1,2). Radiation dose was calculated on a per image voxel basis, considering all available physical interactions. A modern CT scanner was modelled (Revolution GSI, General Electric Medical Systems, WI, USA). Monte Carlo simulations were performed on the patient models using the scanner geometry, the energy spectrum of the X-ray beam and the composition and geometrical characteristics of the beam filtration. Following Monte Carlo simulation of each patient model, an output color-coded image series was generated (Fig. 1). These images depict the normalized to free-in-air CTDI (mGy/100 mAs) dose distribution imparted in the patient's body, in voxel-to-voxel correspondence to the input CT image series.

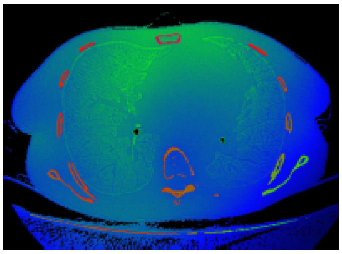

Fig. 1. A color-coded output dose image generated from Monte Carlo simulation.

Organ tissue dose information was extracted from 3D dose distributions through appropriate delineation. To correlate with organ doses, the water equivalent diameter (WED) was measured at the central axial slice depicting heart. WED and organ dose correlation was determined through regression analysis. The results consist of normalized organ doses as a function of WED for 3 kVp values. Normalized organ doses correlated strongly with WED (R^2>0.8 for most cases). Accurate estimation of organ doses from chest CT examinations can be made using correlation equations developed in this study.

Keywords: Chest CT, personalized CT dosimetry, organ doses

ACKNOWLEDGMENTS

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OS4.1 (T4.3-0262)

Analysis of reasons for the multiple scans of paediatric CT examinations in Japan: What causes the different number of scans in disease with the same ICD code

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In recent years, there have been many reports that there is a causal relationship between CT dose and the excess incidence of leukemia and brain tumors in children. These reports show that the relative risk linearly increased with red bone marrow or brain doses from multiple CT scans [1,2]. Although the causal model of radiation exposure would be questioned, why frequent CT scans are taken in a specific child remains unclear. To find the clue to the reverse causation, the reasons why frequent CT scans are taken in a specific child need to be investigated. Our previous study indicated the medical reasons for more frequent examinations were trauma and hydrocephalus, and also showed that 8.5% of CT examinations were done three times or more, in contrast to 91.5% for one (78.8%) or two CT scans (12.7%). This study conducted a survey in the major hospitals for pediatric diseases in Japan, and classified the reasons more frequent examinations using the International Statistical Classification of Diseases and Related Health Problems 10th Revision (ICD-10). We analyzed the reason why the number of examinations was increased for children. The top three reasons for performing CT scans three times or more were hydrocephalus, craniosynostosis, and arachnoid cysts, all of which were related to hydrocephalus. The ranking was 4th brain tumor, 5th neuroblastoma, 7th acute subdural hematoma, and similar to our previous report in a different hospital. We focused on the trauma in children. Examination of pathological conditions in trauma is very urgent, such as a bruise causing localized internal bleeding. Among the traumas, the most frequently examination reason was acute subdural hematoma, and the number of examinations differed depending on the size and condition of the hematoma. As for the interval of examinations, about 40% of all examinations were performed within one week after the first CT examination, and it was found that most of the examinations were conducted within one week after injury. According to this survey, what caused the differences in the number of CT examinations were dependence on the symptoms and conditions even if the characteristics were the same due to the difference in disease or the same disease. This study revealed that frequent CT examinations higher than five scans seemed to have justified reasons. However, further studies will be needed to clarify whether frequent CT examinations higher than five scans with medical reason of trauma would bring brain cancer, as the average dose to brain is 40 mGy at the age of 5 years according to our calculation using WAZA-ARI [3].

***Keywords**: CT examinations, Medical condition, Childhood cancer*

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OS4.1 (T4.3-0495)

Exposure Doses from Pediatric CT in Tunisia: Implementation of National Diagnostic Reference Levels

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The usefulness of CT in the medical management of patients is undeniable and its use continues to grow despite its highly radiating character posing a more delicate radiation protection problem in children because their high radio sensitivity and longer life expectancy than adults.

This work focuses on the determination, by experimentation and simulation, of the radiation doses for a total sample of 916 children, categorized in four age groups (<1,1-5,5-10,10-15 y), underwent the most frequent pediatric CT scans performed in seven different facilities, representing five major geographic regions in Tunisia, and then the development of National Diagnostic Reference Levels of the country. Dose evaluations include $CTDI_{vol}$, DLP, Effective Doses as well as absorbed organ doses. The different pediatric CT protocols, practices as well as the image quality are also evaluated.

Results show large variations in doses between different radiology departments. The proposed national DRLs across all age categories are 6–51 mGy ($CTDI_{vol}$) and 384– 978 mGy.cm (DLP) for Head examinations; 8–16mGy ($CTDI_{vol}$) and 118–579 mGy.cm (DLP) for Chest examinations; and 9–18mGy ($CTDI_{vol}$) and 353–1073 mGy.cm (DLP) for Abdomen examinations (Table 1). Organ doses reach 52 mGy for Eye lens and Brain (Head CT), 62 and 20 mGy for Breast and Thyroid, respectively, (Chest CT) and vary between 3 and 58 mGy for Colon (Abdomen CT).

This is the first pediatric CT dose assessment entirely based on a national pilot study. This study shows that optimizing protection for pediatric CT procedures should be a priority especially within the regional hospitals. The implementation of corrective actions will take place after the initial DRLs. These actions, including recommendations and guidelines to good practice, should be a joint effort of all stakeholders, including health authorities, radiation protection regulators, professional societies and universities using interdisciplinary working groups.

Table 1. Proposed National DRLs ($CTDI_{vol}$, DLP) and effective doses (E) for Head, Chest and Abdomen examinations

	Head			Chest			Abdomen		
Age (y)	$CTDI_{vol}$ (mGy)	DLP (mGy.cm)	E (mSv)	$CTDI_{vol}$ (mGy)	DLP (mGy.cm)	E (mSv)	$CTDI_{vol}$ (mGy)	DLP (mGy.cm)	E (mSv)
<1	25,9	384	1,9	7,8	16,3	6,3	8,8	353	15,0
1-5	37,6	664	2,2	9,8	118	5,6	12,8	485	2,5
5-10	50,7	873	2,7	12,2	330	6,6	16,6	1204	8,5
10-15	50,8	978	2,4	16,3	442	7,7	18,5	1073	10,1

***Keywords:** Pediatric CT, Optimization, Radiation protection*

OS4.1 (T4.3-0479)

Assessment of the feasibility of establishing local diagnostic reference levels for paediatric interventional radiology procedures

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Methods: During 18 months, the type of procedure, patient's weight and age and dose-related data of 279 paediatric interventions were registered in a database, as a result of multidisciplinary teamwork that involved interventional radiologists, radiographers and medical physics technicians. Local DRLs were only obtained when a sample of at least 15 patients could be gathered for a given procedure and weight range [1] and were defined as the third quartile (Q3) of the kerma-area product (P_{KA}) values. As suggested by the ICRP, the Q3 of the fluoroscopy time (FT) and number of digital subtraction angiography (DSA) images were also calculated [2]. Finally, the correlation between P_{KA} and weight was also analysed.

Results: Local DRLs were obtained for these PIR procedures: hepatic/biliary interventions (5-15 kg, 1304 cGy·cm^2; 15-30 kg, 2121 cGy·cm^2), sclerotherapy procedures (15-30 kg, 704 cGy·cm^2; 30-50 kg, 4049 cGy·cm^2; 50-80 kg, 3734 cGy·cm^2) and central venous catheter (CVC) procedures (5-15 kg, 84 cGy·cm^2). The correlation between P_{KA} and weight was weak for sclerotherapy interventions (r=0.34) and just moderate for hepatic/biliary procedures (r=0.61). In the case of CVC (Fig. 1), a higher correlation was found if the fluoroscopy P_{KA} value was normalized to the FT (r=0.85 vs r=0.35).

Conclusions: For the first time in a European country, local DRLs are established for the three most common PIR procedures: sclerotherapy, hepatic/biliary and CVC interventions. The reduced number of paediatric procedures was clearly the main problem encountered, which was aggravated by the use of weight ranges. However, the weak correlations found make it impossible to use alternative metrics that avoid the use of ranges. This poor correlation between P_{KA} and weight is possibly due to the high variability showed in the complexity level. In this regard, an effort should be made to incorporate complexity into the analysis.

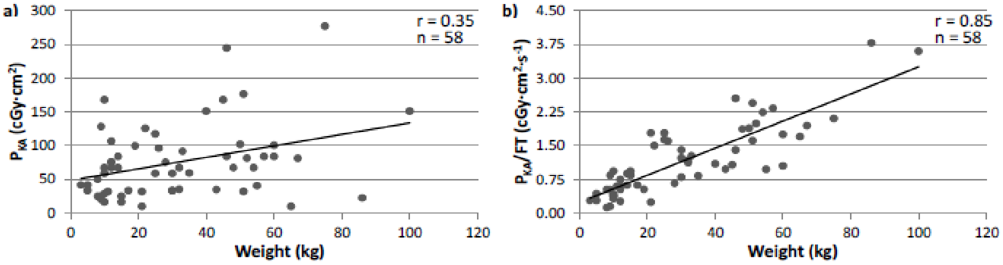

Fig. 1. P_{KA} (a) and P_{KA} normalized to the FT (b) presented as a function of body weight for CVC procedures. The Pearson correlation coefficient (r) and the number of patients (n) are also indicated.

***Keywords**: Paediatrics, Interventional Radiology, Diagnostic Reference Levels*

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OS4.2 (T4.4-0304)

Detection of interfractional range shift in spot scanning proton therapy using gamma electron vertex imaging system

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The proton beam is highly sensitive to uncertainties, and this sensitivity limits the potential benefits of proton therapy over conventional therapies. The reduction of beam range uncertainties may translate into reductions of treatment margin and delivered dose to healthy tissue, and this would likely enhance the clinical benefit of proton therapy. To reduce the range uncertainties, verification of the proton range with prompt gamma (PG) imaging has been pursued. The gamma electron vertex imaging (GEVI) system, a PG imaging device, has been developed for application to spot scanning proton therapy [1]. In this study, the interfractional range shifts in spot scanning proton therapy were retrieved using the GEVI system for various shift scenarios. Interfractional range shifts were realized by repeatedly delivering a same spot scanning proton beam in different shift scenarios. For the treatment planning, we used a single field cubic dose distribution with 2 Gy fractional dose, 50 mm^3 target volume, and total 12 energy layers. The most six distal energy layers were delivered to a homogeneous PMMA phantom, and the GEVI system monitored the range shifts. The range shifts were introduced by covering the beam path with various thick PMMA phantoms (5, 7 and 10 mm) in two ways: global and local shifts; which respectively affect the beam range for every spots and partial spots in each layer. The shift scenarios were evaluated by spot-wise analysis: comparing two spots with and without shift, and statistical hypothesis test: ANOVA to investigate whether each range shifts are significantly different from other introduced shifts, and one-sample t-test to prove whether the measured shift differs from zero. The statistical tests were performed using SPSS 25 (IBM, Armonk, NY) with 5% significance level. To improve prompt gamma statistics, adjacent spots were aggregated where each spot, including itself, is replaced with the sum of spots in each energy layer. In spot aggregation, two-dimensional Gaussian model with 7.8 mm sigma was utilized, weighted by their respective distance from the spot and their delivered dose. In Monte Carlo simulation, Geant4 (ver.10.04.p02) was used. For hypothesis tests, all shift scenarios were statistically distinguishable as results of ANOVAs and were significantly different from zero as results of one-sample t-tests, which indicates that the range shifts can be detected certainly in various conditions using the GEVI system. For all investigated energy layers, the introduced global and local shifts were quantitatively retrieved with accuracies of -0.53, 1.86 mm and precisions (1.5σ) of 0.11, 0.17 mm. The estimated precisions were greatly smaller than the typical safety margin (3% + 3 mm) currently applied in proton therapy, which suggests that the safety margin can be reduced with the application of the GEVI system into proton beam monitoring. This simulation study will be extended to experimental study using a therapeutic proton beam.

Keywords: *Spot scanning proton therapy, Beam range verification, Prompt gamma imaging*

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OS4.2 (T4.4-0314)

New Prototype of Multi-slit Prompt-gamma Camera for Range Verification in Spot Scanning Proton Therapy: Performance Prediction with Monte Carlo Simulation

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In the field of proton therapy, various in vivo range verification methods have been studied for the reduction of range uncertainty, which may lead to less safety margin and more conformal dose delivery. In this context, a multi-slit prompt-gamma camera was developed to verify beam ranges by imaging distribution of prompt gammas emitted along the proton beam trajectory [1]. In spot scanning proton therapy, however, the counting statistics of the existing camera was not enough to measure the ranges of spot beams, making it difficult to reduce the margin considerably. In the present study, to improve counting statistics and the precision in range measurement, a new prototype of multi-slit prompt gamma camera was designed, and its performance was evaluated comparing with the existing camera using Monte Carlo simulation. Compared to the existing camera, the design of the new prototype has differences in the dimensions of slit, collimator, and scintillator. The widths of the slits and the collimators were widened from 2 to 3 mm to acquire more counts on each channel and enhance spatial collimation. The volume of scintillators was increased from $3\times30\times100$ mm^3 to $4\times80\times100$ mm^3 for higher detection efficiency. To compare the range measurement accuracy of the existing and the new camera, Monte Carlo simulations were conducted using Geant4 (ver. 10.04). In the simulations, a cylindrical water phantom (d = 15 cm, h = 25 cm) was irradiated by proton beams, and each camera which was located at 10 cm away from the beam measured the ranges. Simulations were repeated for 10 times for each condition which was varied with beam energy (100, 130, and 160 MeV) and the number of protons in a beam (3×10^8, 1×10^8, and 3×10^7). Figure 1 shows the mean and precision (1.5σ) of the measured ranges for each condition. In all cases, the new prototype had substantially higher accuracy and precision than not only the existing camera but also typical safety margin for proton beam range (3% + 3 mm). Considering that the average number of protons in one monitor unit and one spot are approximately 1×10^8 and 3×10^7 (at National Cancer Center), the counting statistics of the new camera appeared to be at a practical level. Based on this study, the new prototype will be developed and evaluated by an experiment.

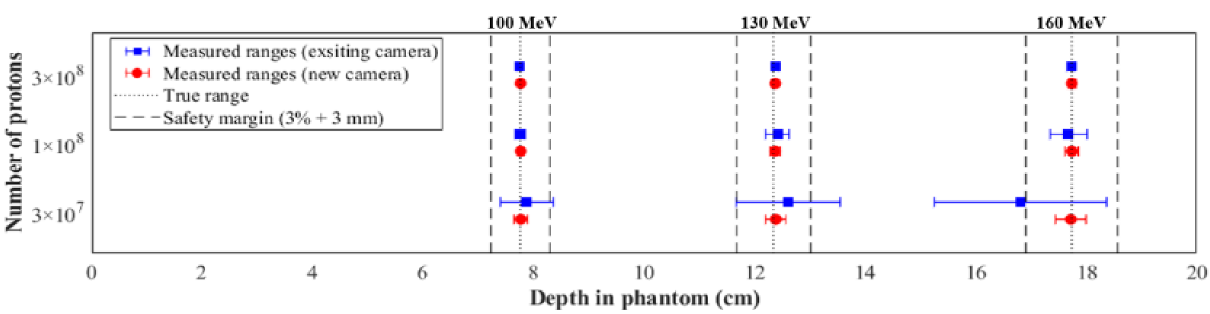

Fig. 1. Ranges (±1.5σ) measured by the existing and the new camera

***Keywords**: Multi-slit prompt-gamma camera, Range uncertainty, Proton therapy*

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OS4.2 (T4.4-0335)

A feasibility study on novel methodologies for patient-specific quality assurance for radiotherapy using deep learning

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The aim of this study is to predict the delivered dose distribution for patient-specific quality assurance (PQA) for radiotherapy without dry-run. We investigated pre-treatment QA of MLC using artificial neural network affordable to time sequence data for prediction of each leaf accuracy. Then a fluence-to-dose network was used to predict the delivered dose distribution to conduct patient-specific intensity-modulated radiation therapy (IMRT) quality assurance. RTpaln dicom file which has information about the treatment plan was converted to Dynalog expected position by in-house program for the estimation of MLC mechanical position before the radiation treatment. Second, we constructed artificial neural network for predicting actual position of MLC reflecting mechanical error of linear accelerator. The neural network was consisted of 5 long short term memory (LSTM) cell (4 gates, 128 neurons) and fully connected layers (1024 neuron) developed using open-source software library for machine intelligence (Tensorflow). Using the predicted MLC information, the expected fluence stack and the actual fluence stack were created from the expected and actual machine parameters, respectively. Finally, the predicted dose distribution was reconstructed for the PQA. A dosimetry using EBT3 films and an ion-chamber array detector (MatriXX) was conducted to evaluate the predicted PQA performance. Comparison of the mean gamma passing rates with a planned dose distribution of the tested cases indicated that these values were within the clinically acceptable levels of accuracy and were, respectively, equal to 98.56% and 96.01% according to the 3%/3 mm and the 2%/2 mm gamma criteria (Table 1).
This work indicates the proposed pre-treatment PQA method based on artificial neural network showed the feasibility to predict delivered dose for quality assurance and patient safety before radiotherapy treatment.

Table 1. The summary of gamma passing rate with 3%/3mm and 2%/2mm gamma criteria for clinical site

Gamma criterion	Clinical site	Brain (%)	Lung (%)	Liver (%)	Prostate (%)	Avg. (%)
3% / 3mm	Proposed method	98.21	97.93	98.74	99.36	98.56
	Proposed method (partial)	97.01	96.34	97.13	97.94	97.11
	EBT3 film	97.85	96.24	96.7	98.14	97.23
	MatriXX	98.4	96.8	97.1	99.8	98.03
2% / 2mm	Proposed method	96.31	95.41	95.69	96.61	96.01
	Proposed method (partial)	93.29	92.17	91.75	94.05	92.82
	EBT3 film	91.44	89.32	90.35	91.43	90.64
	MatriXX	95.37	93.82	94.66	96.91	95.19

Keywords: Dose prediction, Deep-learning, Patient-specific quality assurance

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OS4.2 (T4.3-0341)

Independent 3D Dose Calculation Tool for Image-guided High-dose-rate Brachytherapy

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Fast, volumetric independent dose calculation (vIDC) tool for image guided adaptive high-dose-rate brachytherapy (BTHDR) was developed to facilitate comprehensive dose evaluation beyond a point-based plan decision. The feasibility and clinical practicality of the vIDC was evaluated for previously treated patient cases. Five cervical cancer patients were selected to evaluate dose distributions in BTHDR using a tandem and ring applicator. A fractional dose of 550 cGy was prescribed to 90% of the high-risk clinical target volume (HR-CTV) across 6 fractions. At each fraction, the dwell time re-optimization was performed to meet dose constraints to organs-at-risk (OAR) contoured at interfractional MRI images. The vIDC adopts an updated version of the TG-43 formalism and the same air-kerma rate for iridium-192 with a clinical treatment planning system (TPS). Volumetric dose evaluation using the vIDC was perfromed by an accuracy test at the ICRU reference points, namely point A and B, and rectal points. Dose differences were presented with dose-volume histograms (DVHs) and primary dose-volume parameters such as D_{2cc} for organs-at-risk (OAR). A grid size of 1.0 x 1.0 mm^2 (G-1.0) was selected for dose calculation. To test the effect of grid size at the high-dose gradient, dose was compared at a sparse and a fine grid resolution of 2.5 x 2.5 mm^2 (G-2.5) and 0.5 x 0.5 mm^2 (G-0.5), respectively. The averaged difference throughout an entire volume of dose points was less than -1.79%. When the 1-mm grid resolution was used, the DVHs for the CTV and OAR showed minimal difference between the vIDC and the TPS. While D_{2cc} of OAR showed averaged dose deviation less than 10 cGy, D_{90} of CTV showed the averaged difference of -12.90%, -8.26%, and -6.18% in G-2.5, G-1.0, and G-0.5, respectively. The vIDC was developed as an efficient second-check 3D dose evaluation tool for BTHDR using iridium-192. Volumetric dose calculation using the sparse grid size can affect delivered dose distributions using volume-based dose prescription.

Table 1. Dose-volume comparison for target volumes (HR-CTV and GTV) and OAR (bladder, rectum, and sigmoid), when the different grid sizes were used for 3D dose calculation with the vIDC.

			Target Volumes		OAR		
			Volume [cc] (Percent difference as compared to TPS, %)		Dose [cGy] (Percent difference as compared to TPS, %)		
	Grid Size	Index	HR CTV Mean ± std	Index	Bladder Mean ± std	Rectum Mean ± std	Sigmoid Mean ± std
TPS	1	D_{100}	301.67 ± 68.43	D_{10cc}	243.30 ± 70.81	138.93 ± 63.98	232.63 ± 65.70
		D_{90}	514.80 ± 39.70	D_{2cc}	363.43 ± 102.29	210.53 ± 79.82	347.67 ± 81.67
				$D_{0.1cc}$	514.70 ± 149.84	290.87 ± 119.04	518.40 ± 136.06
vIDC	1	D_{100}	278.23 ± 64.70 (-8.26%)	D_{10cc}	258.67 ± 76.00 -6.22%	134.43 ± 54.38 (-5.70%)	222.70 ± 62.18 (-4.02%)
		D_{90}	486.87 ± 41.17 (-5.43%)	D_{2cc}	392.97 ± 114.24 -7.69%	197.27 ± 70.64 (-5.56%)	333.20 ± 79.21 (-4.15%)
				$D_{0.1cc}$	572.70 ± 175.78 -10.55%	267.43 ± 102.38 (-7.18%)	489.00 ± 128.58 (-5.43%)

OS4.2 (T2.1-0545)

MONITORING ADVENTITIOUS RADIATION EXPOSURE IN RADIOTHERAPY- Radiotherapy Attributable Second Primary Cancers

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Advances in radiotherapy techniques, together with improvements in the early detection of cancer and in supportive care, have contributed to steady gains in the expectation of survival of patients suffering cancers. As a result, the prospective risk of long-term surviving patients to incur 'second primary cancers' attributable to radiotherapy is becoming an important issue particularly for pediatric patients. The prospective risk of such 'second primary cancers' is associated to the radiation exposure incurred in the course of radiotherapy procedures. Most of this exposure is therapeutic but some adventitious exposures are unavoidable. Regulatory requirements for monitoring these exposures are very limited. This paper explores relevant radiation protection issues in radiotherapy and presents some considerations on potential regulatory actions for monitoring and recording exposures incurred in radiotherapy procedures. A document has been prepared which particularly addresses unwanted radiation exposure in radiotherapy, namely not wished or desired exposures that however are incurred unavoidably and unintentionally during radiotherapy procedures; these will be identified hereinafter with the acronym URER (Unwanted Radiation Exposure in Radiotherapy). URERs can be monitored and recorded, either by measurement or by estimation, using dosimetric quantities or suitable proxies. The estimations of URER for individuals and populations can be approached by developing a database of typical dose distributions for different radiotherapy techniques, patient groups and treatment areas and diagnosis. These estimations can be complemented by experimental approaches to dosimetry of non-target and out-of-field exposures. The potential role of biological dosimeter will also be discussed.

Keywords: *second primary cancers, radiotherapy, attribution, monitoring*

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Oral Session T4

OS4.3 (T4.4-1251)

Development of integrated prompt gamma imaging and positron emission tomography system for in vivo 3-D dose verification: a Monte Carlo study

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An accurate knowledge of *in vivo* proton dose distribution is very important for fully utilizing the potential advantage of proton therapy. Among various methods for in vivo range verification, there are two representative indirect methods: prompt gamma (PG) imaging and positron emission tomography (PET). In this study, we proposed a PG-PET system that combines the advantages of those methods and presents detector geometry and background reduction techniques optimized for the PG-PET system. The characteristics of secondary radiations emitted from a water phantom after interacting with 150 MeV proton beam were analyzed by Geant4.10.00, and the 2D PG distributions were obtained and assessed according to different detector geometries. As the background reduction techniques, energy window (EW), depth-of-interaction (DOI), and time-of-flight (TOF) techniques were proposed. For the performance evaluation of the PG-PET system, 3D dose distribution in water phantom caused by two proton beams of 80 and 100 MeV energies was verified using 16 optimal detectors. As the results of the optimization study, the 200 mm thickness of the parallel-hole tungsten collimator with 8 mm pitch and 7 mm hole width, and the 30 mm thickness of GAGG scintillator were determined. If the DOI technique was applied for data processing, 3-7 MeV was an optimal EW; the detector performance was improved by about 38% compared with that when applying only 3-5 MeV EW. The TOF technique was demonstrated as the most powerful background reduction technique. The reconstructed 3D PG and PET distributions in the water phantom well estimated the proton dose distribution. In the current study, we confirmed that the PG distribution could be measured by just combining the 2D parallel hole collimator with the PET detector module. In the future, an experimental validation study of the PG-PET system will be performed with two detector modules.

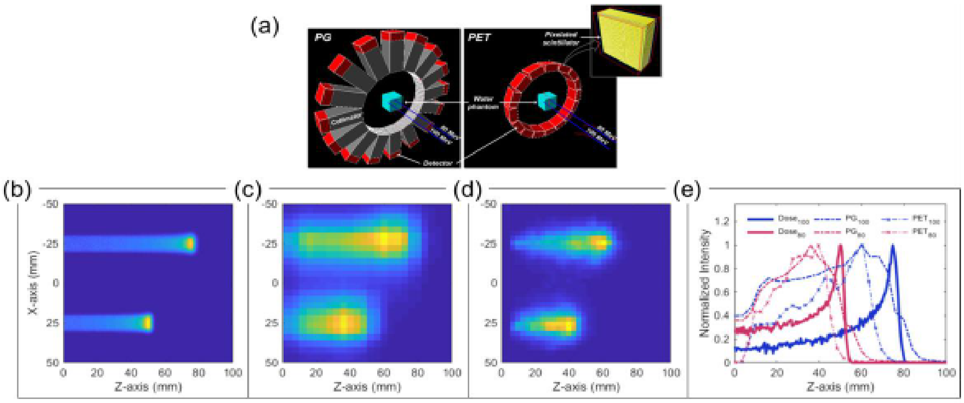

Fig. 1. (a) Geant4 simulation model of the PG-PET system, (b) Proton dose distributions, (c) Prompt gamma distribution, (d) Positron emitter distribution reconstructed by the MLEM algorithm, (e) Comparison of depth-directed image profiles to assess the capability of proton range verification

Keywords: proton therapy, prompt gamma imaging, positron emission tomography, in vivo dose verification, Monte Carlo

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OS4.3 (T4.4-0221)

Three Years of Experience Treating Patients with Lutathera® (Lutetium, Lu-177, dotatate)

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Lutathera® is indicated for the treatment of somatostatin receptor-positive gastroenteropancreatic neuroendocrine tumors (GEP-NETs) and was approved by the United States Food and Drug Administration for use as a commercially available therapeutic agent on January 26, 2018. Fox Chase Cancer Center first started treating patients with Lutathera® in December 2016 under a clinical trial and progressed to offer Lutathera® to patients as a treatment option in April 2018 and has administered Lutathera® 102 times as of September 2019. The radiation safety considerations for the use of Lutetium 177 (Lu-177) include radioactive materials licensing, staff training, treatment room preparation/decontamination, and radiation safety precautions for patients. Lessons learned include verifying that the toilet is not leaking in the treatment room prior to therapy, allowing patients to remain in their own clothing rather than changing into hospital gowns, requesting that all patients use a seated position when using the bathroom, patients should drink a lot of fluids but they can drink too much, we need to be aware of holiday schedules for international therapy agent production facilities, treatment room preperation is different for incontinent patients, patients can be discharged the same day as therapy under United States Nuclear Regulatory Commission regulations, and gamma well counter efficiency for Lu-177 is very low. Lessons learned are offered to help other radiation safety offices as more medical centers begin to offer Lutathera® to their patients.

***Keywords**: Lutetium 177, Lutathera, Radiation Safety*

OS4.3 (T4.3-0412)

Using Y-90 Medical Event Reports to Inform and Improve Radiation and Patient Safety

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Y-90 microspheres are indicated for the treatment of hepatocellular carcinoma and metastatic colorectal cancer and have been used to treat patients for over 20 years. Over the past 4 years, incidents involving Y-90 are reported to United States regulators more frequently than any other radioactive materials therapy. Incidents that have been reported to the United States Nuclear Regulatory Commission were reviewed and were classified into different apparent root causes. We will share ideas for procedures, processes and training to address identified issues could lead an event. Physicians, nurses, radiation safety staff and vendor representatives provided input from their perspective for solutions. Prevention is driven by experience. Lessons learned are offered to help other practices as they establish or improve their Y-90 programs.

***Keywords**: Yttrium 90, TheraSphere, SIR-Sphere, Radiation Safety*

OS4.3 (T4.B-0650)

Effective risk-benefit dialogue with patients, public & professional staff – communication of the justification of whole body F18-FDG PET/CT before treatment of pregnant patient with Hodgkin Lymphoma malignancy

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In pregnancy, Hodgkin Lymphoma is one next most common type of tumour after breast cancer. Our institute has more than 30 years of experience with successful treatment of pregnant patients including 147 with Hodgkin Lymphoma. Over time, the treatment approach has developed due to technological and clinical changes. Now a days the ISRT (Involved site radiation therapy) has replaced MANTEL fields approach. The ISRT requires a PET/CT examination before treatment to accurate delineate CTV after chemotherapy and monitor clinical response for chemotherapy during treatment. The F18-FDG PET-CT procedure is commonly used for the patients, but for a pregnant patient the justification of the procedure is still a very sensitive topic, and this relevant examination might not be preferred due to staff concern about the foetus radiosensitivity and the legal aspect of the examination.

The F18-FDG PET procedure on Philips Gemini TF scanner has been under evaluation, all relevant parameters from our procedure and our practice have been taken into account. Due to available TOF (time-of-flight) option on PET/CT scanner a 3.7 MBq /kg optimized dose is used. Total foetal whole body dose associated with the F18-FDG PET/CT examination has been estimated as 2.5 cGy which is 25% of the safe limit introduced by the ICRP (10cGy). This dose includes 0.6 cGy coming from optimized 18-FDG PET, and around 1.7 cGy coming from the whole body CT. Theoretically, the additional risk of childhood cancer associated with this examination, using LNT approach, is 0.15% (0.3% background), resulting in total 0.45% risk (ICRP pub84).

The F18-FDG PET/CT is a 'must' procedure during Hodgkin lymphoma treatment. Benefits for the mother coming from proper diagnosis and therefore further minimizing the irradiated volume significantly overweighs potential risk for foetus. This means that the F18-FDG PET/CT for pregnant patient with Hodgkin lymphoma is justified and should be performed. The Basic Safety Standards and the local law requirement do not forbid PET/CT examination as long as it is clinically justified, optimized and communicated to the pregnant patient prior treatment. Effective risk-benefit dialogue with the patient and the family, the public and the professional staff, should include above mentioned facts. The final clinical decision concerning the type of medical examinations (diagnosis and treatment) depends on trimester of the pregnancy, histopathological diagnosis, stage of disease, fetus position, treatment benefits, and the reproductive capacity of the patient (fertility), patient / father of the child / family wish and will; and should take into account also ethical aspects.

Keywords: *Hodgkin Lymphoma in Pregnancy, PET/CT, communication of benefit & risk*

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OS4.3 (T3.D-0265)

Using SAFRON to Improve Safety And Quality in Therapeutic Nuclear Medicine

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The role of nuclear medicine therapeutic treatments is established and expanding, thanks to new radiopharmaceuticals that allow targeting malignancy with a high level of selectivity, releasing only limited radiation dose to healthy tissues, and considering the potential of the theragnostic approach. In the majority of cases, these treatments are well tolerated by patients and have limited deleterious effects. Nevertheless, in a complex, highly structured discipline like nuclear medicine, there remains the probability of procedural errors, unexpected events or unforeseeable aspects that may lead to inappropriate delivery of the therapy, undesired patient exposure or other consequences potentially harmful to patients. The rapid availability of these new treatments have provided positive benefit but safety concerns may have been overlooked in the promotion of the benefits of the therapy. Some significant reports have been presented in the current scientific literature.

Incident reporting systems are a tool in health institutions to monitor unexpected events, incidents and close calls, activate corrective actions and provide useful information to the clinicians to prevent repetitions. The IAEA introduced SAFRON in 2012, a web-based system for incident reporting in radiotherapy. A meeting on prevention of incidents in nuclear medicine held in May 2018, with participation of an international, multidisciplinary panel of experts, produced a series of recommendations including extending SAFRON to incidents in nuclear medicine therapeutic applications. These include therapies with radiopharmaceuticals and radioactive medical devices such as SIRT. A specific task group convened in November 2018, reviewed existing reports and experiences, and develop the parameters of the application, define the scheme of incident models, the modality of registration and the possible safety barriers. The new addition to the SAFRON platform was developed in late 2019. The introduction of data is easy and anonymous, guaranteeing the privacy of patients and reporting institutions, but nevertheless making possible for professionals in the field to obtain relevant information on incidents, their modality, consequences and possibilities for mitigation, and contributing then to diffuse knowledge and learn from previous lessons.

Participating facilities will be able to use SAFRON NM as their local incident learning system as well as contributing to an international incident learning system looking for opportunities to improve the safety systems and reduce potential errors. SAFRON NM is a state of the art, safe and effective incident learning tool available to the nuclear medicine community to foster prevention of incidents and improve the safe administration of therapeutic radionuclides.

OS4.3 (T3.D-0265)

Using SAFRON to Improve Safety And Quality in Therapeutic Nuclear Medicine

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Keywords: *Radionuclide therapy, Safety, Incident learning systems*

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OS4.4 (T4.1-0092)

Retrofitting of Existing Structural Shielding in Diagnostic Radiology facilities

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Structural radiation shielding calculations for diagnostic x ray facilities is most commonly performed using the recommendations of National Council on Radiation Protection and Measurements Report No. 49 which continues to be the primary guide for diagnostic x-ray structural shielding design for a while. Many changes have occurred over the years that have caused the NCRP Report 49 calculation methodology to become essentially obsolete in that it did not address technology advances in Radiology. The methodology was remedied with the release of NCRP Report No. 147 which recommended lower dose constraints as a design target goal. Consequently, retrofitting of existing structural shielding become inevitable and unavoidable.

In general, optimization can be accomplished mainly through an optimal planning and construction of the structural shielding. Therefore, FANR developed software for shielding calculations based on NCRP report 147 that was validated by IAEA independently, aims to provide a tool to assist medical physicists and other radiation protection professionals in the planning and design of new x-ray facilities and in the remodeling of existing facilities in order to comply with the imposed safety requirements.

The corrections or additions after facilities are completed and existing are usually expensive. In addition, most institutions may undertake substantial changes by entering into new businesses or services, or to purchase new x-ray modality, or convert the use or occupancy of existing x-ray rooms or an adjacent areas. However, redesign of existing shielding as accurate as possible, taken into account the cost of shielding, use of alternative additional shielding materials, apply the ALARA principle and consider monetary cost-benefit.

In order to obtain as accurate as possible the equivalent and adequate shielding required and avoiding the extensive costs to achieve, at the same time, demonstrate compliance with imposed regulatory requirements. FANR software for shielding calculations has been used to collect a large set of shielding thickness data for different shielding materials (concrete, lead, steel, Plate Glass and Gypsum) and for wide spectrum of x-ray modalities at diverse setting and assumption. In addition, a conservatively safe approach in specifying radiation barriers has been applied.

The collected data have been carefully analyzed and evaluated. The analyzed data shows a constant proportional between different shielding materials for all investigated modalities at different exposures. The obtained relationship which is representative by the proportional or equivalency factors (will be presented as table) between diverse shielding materials can be used effectively by both, of the regulators and other radiation protection professionals to estimate and evaluate an accurate thickness for any additional shielding materials needed to achieve goals of the radiation safety when other shielding materials would be used or when the existing shielding that will be utilized as part of the design. The shielding calculation tool served as an effective method to demonstrate compliance with the regulations when used by both the applicants and licensees.

OS4.4 (T4.2-0428)

An innovative system for preventing inadvertent exposition of staff outside the operating theater

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Context: Mobile C-arms move throughout operating rooms of the operating theater. Being designed to move between rooms, they are not equipped with relays to retrieve the exposition information and export it outside the room. Therefore, no light signaling is available outside the room to warn the X-ray emission for staff. Inadvertent exposition of staff outside the operating theater is a real problem for radiation protection. The French standard NFC 15-160 require that: (1) access to any room containing an X-ray emitting device must be controlled by a light signage so that it cannot be inadvertently crossed, and (2) setting up an emergency button to stop the X-ray emission.
We present an innovative system that we developed to meet these requirements.

Materials and methods: The system is composed of two communicating boxes:
The "DetectBox" is to be installed inside the operating theater. It identifies the various operation states of the C-arm by analyzing its power supply signal. The DetectBox communicates (in wireless mode) with the second box (AlertBox).
The "AlertBox" can operate in socket or battery mode and is to be installed outside the operating theater. It detects and reports the state of the C-arm by emitting a real time light signal. This latter can have three different colors: Red when the C-arm is emitting X-rays, Orange when it is Powered On but does not emit X-rays, and Green when it is Powered Off.
The two boxes communicate on a radiofrequency link exclusively carried out in the 'Industrial, Scientific and Medical (ISM)' frequency bands and allows the coexistence of several on-site warning systems without communication conflicts (interference).
Taking into account the complexity of performing electrical works in the operating theater (for reasons of hygiene and continuity of medical care), this patented system and with the size <10 cm^2, works in complete safety without any intrusion in the mobile C-arm and does not require specific electrical installation work. The system is equipped with emergency button that stop X-ray emission.

Results: The system has been tested, and it has been shown that: it detects X-rays having both high and low energy (50 – 150 kVp), high and low photon flow (0.5 – 200 mA: even when emitted for a very short time (<1 ms)), Probability of false detection < 10-5, it operates under all acquisition modes (continuous, pulsed, fluoroscopy mode, image mode, subtraction and movie mode), it is compatible with all C-arm models and brands.
We have also tested the communication between the 2 boxes (DetectBox and AlertBox) in several conditions: (1) Unleaded room, (2) leaded room, and (3) rooms with particular configuration (sas, great distances, concrete walls, 3 mm of lead). The result of these last tests was positive.

Conclusion: This innovative system is a reliable tool to alert the staff outside the operating room for X-ray emission and insure their radiation protection.

OS4.4 (T4.2-0461)

Design of radiological area monitoring in compact proton therapy centers (CPTC)

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The advantages of proton therapy (PT) in some treatments again cancer have resulted in a significant expansion of proton therapy centers (PTC) around the world, with almost one hundred placed mainly in develop countries. The current trends are to build small compact and standard facilities, along with the renovation of the former multiple room centers (MPTC). Compact Proton Therapy Centers (CPTC) usually have one treatment room, comprising latest technological advances in radiotherapy. In the interactions of protons used in therapy there is a huge production of stray radiation, neutron and gamma mostly, therefore optimal design of radiological area monitoring must be developed and carried out in commissioning phases.

The aim of this work was to design the operational radiological protection in a compact proton therapy center (CPTC) by selecting the radiation detection devices and the REM-meters for high energy neutrons, as well as its location in the center, to develop the radiological monitoring of the area, with full guarantee and compliance of the limits of doses for professionals, clinical staff and the general public. Several models of the radiation sources and materials of facility were simulated, starting from a conservative assumption, followed by more realistic models. Evenly, the neutron fields and spectra present in the installation were characterized selecting the most appropriate radiation measurement device in each location. This work is included in the project *Contributions to Shielding and Dosimetry of Neutrons in CPTC*.

Keywords: *Compact proton therapy centers, Area monitoring, REM-meters*

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OS4.4 (T4.2-0635)

Distributions of Scattered Radiation During Fluoroscopic Exposure from Over-couch and Under-couch X-ray Tube Geometry

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In fluoroscopy examinations, relatively high level of occupational dose is delivered and the interest has recently focused on the occupational dose to the lens of the eye. We aimed to evaluate distributions of scattered radiation when assuming an endoscopic retrograde cholangiopancreatography (ERCP) procedure under three types of geometry: over-couch X-ray tube and under-couch X-ray tube with and without a 0.35-mmPb radioprotective curtain SA-35 (Kuraray Trading, Tokyo, Japan). An X-ray fluoroscopy system Ultimax-i (Canon Medical Systems, Otawara, Japan), which can switch the X-ray tube location between over-couch and under-couch, was used. Under three types of geometry, ambient dose equivalent $H^*(10)$ rates at 150 cm above the floor were measured at 92 points with a survey meter ICS-331B (Hitachi, Tokyo, Japan). A whole-body phantom PBU-60 (Kyoto Kagaku, Kyoto, Japan) was placed on the couch, and exposure parameters were adjusted according to the operation of automatic brightness control. The $H^*(10)$ rates for the under-couch X-ray tube geometry were 0.0-98.5% lower than those for the over-couch X-ray tube geometry at all the measurement points (Fig. 1(a) and 1(b)). The $H^*(10)$ rates for the under-couch X-ray tube geometry with the curtain were 2.4-87.3% lower than those without the curtain at 88 of 92 measurement points (Fig. 1(b) and 1(c)). At the endoscopist position, the $H^*(10)$ rates were 5650, 106, and 81 µSv/h for the geometry of over-couch X-ray tube, under-couch X-ray tube without the curtain, and that with the curtain, respectively. Under-couch X-ray tube geometry is strongly recommended during ERCP procedures, and radioprotective curtains are effective for reducing scattered radiation around operators' positions.

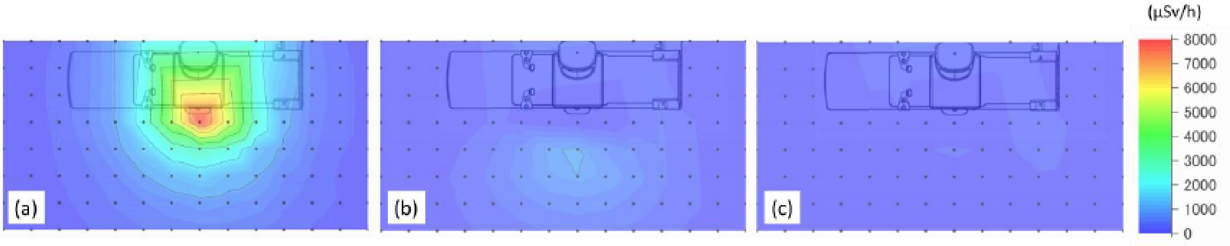

Fig. 1. Distributions of $H^*(10)$ rate: (a) over-couch X-ray tube geometry, (b) under-couch X-ray tube geometry without the curtain, (c) under-couch X-ray tube geometry with the curtain

***Keywords**: Occupational dose, Scattered radiation, Fluoroscopy*

ACKNOWLEDGMENTS
We would like to thank Ms. Yoko Araki, Mr. Kentaro Naka, and Mr. Shinnosuke Inomata of Canon Medical Systems for their technical and material support.

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OS4.4 (T4.1-0688)

How radiation protection issues influence the design of a clinical facility for accelerator-based Boron Neutron Capture Therapy

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Boron Neutron Capture Therapy (BNCT) is hadrontherapy for the treatment of tumours, in which the patient is irradiated with low energy neutrons after the administration of a boron-10 drug that accumulates in malignant cells. The thermal neutron capture in B-10 produces two high-LET particles, which lose energy in a path comparable with a cellular diameter, causing irreversible damages to the DNA. The use of suitable formulations, able to concentrate boron selectively in malignant cells, allows a lethal effect for tumour while sparing the surrounding healthy tissues. This selective effect makes BNCT a potential therapeutic option for disseminated, infiltrated or non-operable tumours. A fundamental branch of BNCT research concerns neutron sources, since typical ones, research nuclear reactors, are not easily adaptable to the needs of a clinical treatment. Is thus becoming increasingly relevant the development of BNCT neutron beams from accelerators, easier to install and maintain in healthcare settings. Worldwide, projects are ongoing to install accelerator-based BNCT clinical facilities with different technologies. The context of this work is an Italian BNCT clinical facility based on a proton accelerator, producing neutrons through the (p,n) reaction on a beryllium target. This work presents experimental and computational studies on environmental and patient out-of-beam dosimetry, relevant for the implementation of an accelerator-based BNCT clinical facility. Experimental measurements include irradiation of samples at the research nuclear reactor in Pavia, and Neutron Activation Analysis with gamma spectrometry. Calculations regarding the quantities of interest were performed by simulating a clinical irradiation in the facility, using the Monte Carlo transport codes MCNP6 and FLUKA. The studies carried out concern the investigation on the neutron activation of materials present in the Beam Shaping Assembly (the ensemble of materials to moderate and collimate the neutrons produced at the target) and in the treatment room, and the calculation of the dose distributions in the room and in the patient organs. The obtained results are intended to produce operative guidelines for the construction and preparation of the treatment room, and for the use and maintenance of the equipment exposed to neutrons. Furthermore, the work shows how these evaluations influence the design of the beam itself. In fact, literature on the evaluation of neutrons sources for BNCT often overlooks this perspective, focusing only on the physical characteristics of the beams, without considering the peripheral dose, the activation of surrounding materials and the ambient dosimetry. Instead, when designing a BNCT facility from accelerator, the optimization of the shielding, the evaluation and minimization of the dose received by the organs outside the treatment area, and the control on the activation of the irradiated materials are closely interconnected. All these aspects are critical not only to design the facility, but also to manage it during its operation. These studies are in line with the present scientific debate on how to establish the clinical potential of a neutron beam for BNCT, and are particularly relevant as the accelerator technology is now spreading BNCT as an accessible radiotherapy treatment in many institutions.

Keywords: *accelerator-based BNCT, neutron activation, dosimetry*

PS4 (T4.4-1251)

Development of integrated prompt gamma imaging and positron emission tomography system for in vivo 3-D dose verification: a Monte Carlo study

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An accurate knowledge of *in vivo* proton dose distribution is very important for fully utilizing the potential advantage of proton therapy. Among various methods for in vivo range verification, there are two representative indirect methods: prompt gamma (PG) imaging and positron emission tomography (PET). In this study, we proposed a PG-PET system that combines the advantages of those methods and presents detector geometry and background reduction techniques optimized for the PG-PET system. The characteristics of secondary radiations emitted from a water phantom after interacting with 150 MeV proton beam were analyzed by Geant4.10.00, and the 2D PG distributions were obtained and assessed according to different detector geometries. As the background reduction techniques, energy window (EW), depth-of-interaction (DOI), and time-of-flight (TOF) techniques were proposed. For the performance evaluation of the PG-PET system, 3D dose distribution in water phantom caused by two proton beams of 80 and 100 MeV energies was verified using 16 optimal detectors. As the results of the optimization study, the 200 mm thickness of the parallel-hole tungsten collimator with 8 mm pitch and 7 mm hole width, and the 30 mm thickness of GAGG scintillator were determined. If the DOI technique was applied for data processing, 3-7 MeV was an optimal EW; the detector performance was improved by about 38% compared with that when applying only 3-5 MeV EW. The TOF technique was demonstrated as the most powerful background reduction technique. The reconstructed 3D PG and PET distributions in the water phantom well estimated the proton dose distribution. In the current study, we confirmed that the PG distribution could be measured by just combining the 2D parallel hole collimator with the PET detector module. In the future, an experimental validation study of the PG-PET system will be performed with two detector modules.

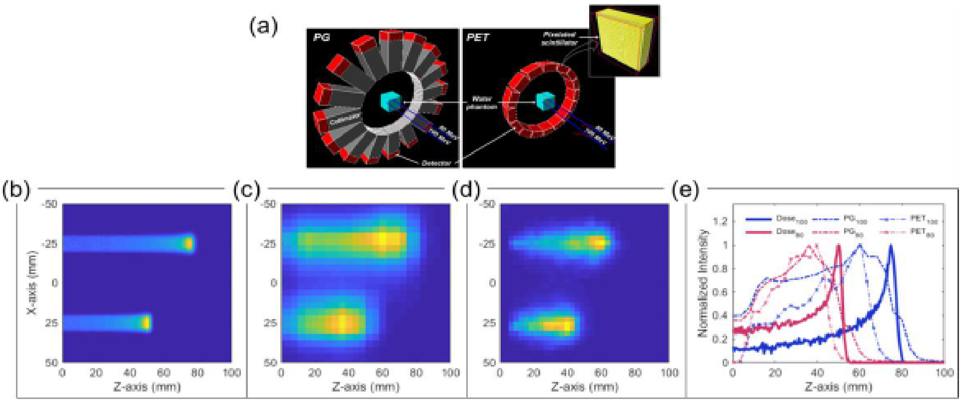

Fig. 1. (a) Geant4 simulation model of the PG-PET system, (b) Proton dose distributions, (c) Prompt gamma distribution, (d) Positron emitter distribution reconstructed by the MLEM algorithm, (e) Comparison of depth-directed image profiles to assess the capability of proton range verification

Keywords: proton therapy, prompt gamma imaging, positron emission tomography, in vivo dose verification, Monte Carlo

ACKNOWLEDGMENTS
This research was supported by the Nuclear Safety Research Program through the Korea Foundation Of Nuclear Safety (KoFONS) using the financial resource granted by the Nuclear Safety and Security Commission (NSSC) of the Republic of Korea (No. 1803027).

OS4.5 (T3.3-0565)

Radiation protection program evaluation in the orthopedic theater of Tunisian Kassab Institute

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Introduction

During the last few years, surgical techniques under brightness amplifiers have increased considerably. The evaluation of the radiation protection organization in the operating room of the Mohamed Kassab orthopedic institute of Tunis, was carried out at the request of the staff which is worried about x-rays exposure from fluoroscopy, to verify that the working conditions are in accordance with the standards and the Tunisian law of Radiation protection.

Materials and methods

A checklist has been prepared in accordance with the recommendations of the evaluation guides of the International Atomic Energy Agency (IAEA).

Results

The answers to the checklist gave the following results. The referent person, was the supervisor of operating room. Regarding the design of the premises, it is authorized by the Ministry of Health and the national auhority in radiation protection and conforms to the radiation protection standards (plan, sealing, concrete) in order to maintain traceability of administrative documents. For the signaling of the regulatory zones (ZC, ZS, ZP), they are not performed and not reported. Although the light signal is present above the door of each fluoroscopy room, it is not functional. We move to the means of protection of the workers against X rays which are either collective or individual. For the collective protection there is only a mobile lead screen and for the individual protection there is a gown and a thyroid cover. Concerning the workers which are exposed to radiation, they are not provided with individual dosimetric monitoring. Even, for patients there is no optimization of the received dose. In this context, information is posted on protection against ionizing radiation with special medical supervision for personnel exposed to these rays, but there is no radiation protection training program for this user staff. Finally, there is no radiological incident recorded and for the quality control there is no traceability for preventive maintenance as well as the RX tube, but the quality control is carried out for the curative maintenance.

Conclusion

Establishing a radiation protection culture on orthopedic operating room is imperative. Training is still one of the main elements in radiation protection. Person competent in radiation protection should be useful in the context of our evaluation.

OS4.5 (T3.D-0253)

Building a radiation safety culture at the Academic Hospital Paramaribo: Crossing the bridge from unknown territory

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Radiation safety education in healthcare doesn't benefit from a one size fits all principle. For the past years, education and training has been focused on hospital employees who are actively using ionizing radiation in their daily practice. Less focus has been placed on other hospital employees who, as a result from their duties may be exposed to ionizing radiation.

A complete overview was made of all hospital employees who are directly or indirectly involved in working with ionizing radiation. Based on their level of involvement and prior training in radiation protection, a training program was developed which included different knowledge levels in radiation protection. Results showed that 14 different groups could be identified, belonging to either category I (those who have received radiation protection training as part of their study program), category II (those who are directly involved in ionizing radiation but have not received formal training) or category III (those who closely work in or around areas where ionizing radiation is being used). Included in the latter group were employees involved in decision making processes, human resource management, technical maintenance and housekeeping and employees at the Quality department. Lack of knowledge in radiation protection may lead to fear which in turns may cause unnecessary absence of employees which is why a session was planned for the labor union.
Since many new employees don't receive instructions on radiation safety, an induction training video was created which highlighted the different principles for radiation protection. The video introduces Radiation Roy (healthcare workers), Radiation Rita and Randy (pregnant and pediatric patient) and Radiation Ruby (public) as the Radiation Family. In the video Radiation Roy switches outfits reflecting the different groups of employees in the hospital and shares information on radiation protection. A three-letter principle was developed per group: Doctors: CPR (Care, Protocol, Regulation: another way of describing justification, optimization, dose limits), nurse supervisors (TLC: Train students and other colleagues, Listen to patient complaints which may be resulting from radiation and Check that every inadvertent event of the patient is documented), health technology engineers (3L: Look for signs, Listen to safety instructions, Learn about radiation), HSEQ personnel and managers (ABC: create Awareness amongst your personnel, Build bridges to your personnel to discuss concerns, Check adherence to radiation safety policy), HSEQ and quality managers (quality policy should include safety measures for 3P's (Patients, Personnel and Public), labor union (3C: Communicate concerns existing amongst members, Collaborate with management to Create and sustain a safety culture), housekeeping (See that there is no radiation present based on radiation signs, Hear no prohibition to enter the room after asking for permission and Speak if you have safety concerns), radiographers (3F: Fast, Far, Fence as another way to describe time, distance, shielding).

Questionnaires revealed that the participants of the training viewed the content as informative and helpful and that knowing how to protect themselves against unnecessary radiation adds weight to their feeling of safety. The video will further aid in the establishment of an effective radiation safety culture.

OS4.5 (T2.2-0413)

Radiation Protection following release of Patients after radioiodine therapy in Madagascar

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Patients treated with radioiodine present a radiation hazard and precautions are necessary to limit the radiation exposure of members of the public and people with whom the patients may come into a contact. Exposure of members of public controlled by keeping the patient in hospital until the retained activity reaches a specific limit. In Madagascar, the patient is released when the dose rate at 1 meter is 40 µSv/h, or a retained activity of 800 MBq according to the European Commission [1]. Then, instructions for limiting exposure to the relatives and the public are given to patient. This paper presents suggestions to the patient's family and the members of the public following a therapeutic administration of a radiopharmaceutical to a patient. For this purpose, simulations were performed for estimating suitable period of restriction during which close contact with the patient should be disallowed and limited. These simulations are based on the French working Group [2,3,4] and the current recommendations of ICRP [5, 6]. Several contact patterns were tested: public transportation, sleeping with partner and close contact with children. The dose constraints were based on calculation by placing permanently dosimeters at the waiting room of the Nuclear Medicine Laboratory of Antananarivo. Thus, the value of 0.3 mSv was used in this work. Simulation was based on the following assumptions. The dose received by an individual from a patient is governed by the rate of iodine and the time spent in close proximity with the patients. In this work, the administration of radioiodine was performed with the therapeutic doses of iodine-131 of 1850 MBq and 3700 MBq. After releasing the patients, the following restrictions were advised. Patients released from hospital could travel in a public car up to 24 minutes by keeping the distance of 0.5 meter from other people. Patients should remain off work for 2 days. Partner should avoid close contact to the patient less than 1 meter and sleep apart for 4 days. Contact with children should be restricted according to their age, 7 days for children under the age of 3 years, and 4 days for children that ages are ranging from 3 to 10 years. The results of this work will provide some practical guidelines to medical professionals involved in release of patient treated with radioiodine therapy.

***Keywords**: Radioiodine therapy, Patient, Restriction*

ACKNOWLEDGMENTS

This work was supported by the International Atomic Energy Agency through the regional project for African Countries on "Strengthening Radiological Protection of Patients and Control of Medical Exposure".

OS4.5 (T2.4-0107)

Local Diagnostic Reference Levels for Digital Mammography: Two Hospitals Study in Northwest, Nigeria

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Optimisation technique remains a key factor in reducing radiation dose to women during mammography procedure. Thus, in this study, local mammography DRL is established for the purpose of achieving patient dose reduction and acceptable image quality based on the ALARA principle. A total of 140 patient data for cranio-caudal and medio-lateral oblique projections were collected and analyzed. Demographic, exposure and dose information was extracted from the digital readouts of the DICOM header stored in the mammography workstation. Exposure parameters and dose data recorded were as follows: compressed breast thickness, kVp, mAs, exposure mode, and MGD values. The result was analyzed using SPSS v.16.0, and DRL was established at the level of 3rd quartile value based on the ICRP guidelines. The established DRL for CC and MLO was found to be 2.31mGy, which compared well with most values reported in the literature, and also, within the European recommended dose levels. Furthermore, the CBT and age were found to correlate positively with the MGD. However, the MGD for manual exposure mode is significantly higher compared to that of the automatic optimisation parameter mode. In conclusion, the locally established DRL align with the recommended European DRLs which is an indication of good local practice. However, there is a need for continuous dose monitoring and image quality assessment in accordance with the ALARA principle to consistently maintain lower patient exposure.

Keywords: Mammography, DRL, MGD

OS4.5 (T8.1-0467)

Managing the magnetic protection of workers in Magnetic Resonance Imaging

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Introduction: In the 'Magnetic Resonance Imaging (MRI)' department, all workers involved in preparing the patient, setting it up, tunnel cleaning, ... are likely to be exposed to 'ElectroMagnetic fields (EMF)' emitted by the MRI device. Exposure to EMF can cause adverse radio-biological effects to workers. The purpose of this study is to propose an organizational process to manage and control EMF risks.

Materials and methods: The study was conducted at seven MRI departments using machines with 1.5 and 3 Tesla magnetic fields. We assessed the exposure of each one by measuring the two electromagnetic fields (static and dynamic) at different distances from the MRI machine both inside and around the examination room. Measurement values were compared with British and American references (those of the UK's 'Medicines and Healthcare Regulatory Agency (MHRA)' and the 'American Radiology Society (ACR)').

Results: Following the results of EMF measurements and their comparison with the recommendations of learned societies, a zoning system that adapts to needs of different MRI services across the country has been proposed. In effect, three risk areas have been identified within the MRI services. This has led to the development of a good practice guide related to the magnetic protection of MRI workers.

Conclusion: The guide established by our study is a standard that allows MRI workers to protect themselves against the risk of electromagnetic fields.

Oral Session T4

OS4.5 (T2.1-0194)

Entrance Dose measurements for patients undergoing X-ray examination, case of private hospital in Madagascar

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Conventional radiology is one of the most used techniques for patient diagnosis in Madagascar. The purpose of this work is to establish a database on the dose received by patients during radiography examinations, to provide methods for optimizing the exposition doses to the patients. For that, a study of entry dose (ED) for some patients undergoing X-Ray examination in one private clinic in Madagascar has been done. During these measurements, examination parameters (high voltage, filtration, field size and source skin distance) for a large number of patients undergoing a different examination were collected and through the output of the X-Ray machine, Entry dose for each patient has been calculated. The realized study established that the mean value for all the realized exposition was 2 mGy, with a maximum of 9.46 mGy corresponding to a Blondeau examination and a minimum of 0.05 mGy for a finger examination. The obtained values can be used for the implementation of the national Diagnostic Reference Levels (DRLs) for some specific examinations and can be a real asset to enhance the radiation protection programme for patients undergoing X-Ray examination at national level.

Key words: Entrance dose, X-ray examination, Optimization

OS5.1 (T5.3-0149)

Assessment of the alpha risk after a fuel dissemination in the Reactor Cooling System of French PWRs using the OSCAR V1.4 Code

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In a nuclear power plant, the alpha risk management is a great concern, due to the adverse effect of alpha emitters in case of incorporation and the primary circuit contamination persistence after a fuel dissemination.
In order to prevent alpha emitter contamination of workers during maintenance operations, EDF implemented a graded approach regarding the alpha risk management. This approach is dedicated to identify as early as possible the maintenance operations with alpha risk and enable the implementation of adapted prevention means (collective and personal protections) in the field.
The first step of the approach is the follow-up of the primary coolant activity, especially the iodine 134 specific activity, which is a good indicator of fissile material presence under neutron flux and thus of a fuel dissemination. In case of a fuel failure (without any dissemination), the iodine 134 is not detected due to its short half-life. Complementarily to the gamma spectrometry measurement performed by the chemistry staff of the NPP, simulation tools can be effective to provide useful information and trends regarding the alpha contamination of the primary circuit, and this for several cycles after the dissemination occurred.

In this paper, we will describe how the OSCAR code can be used to simulate the behavior of alpha emitters in the reactor cooling system of a PWR. The OSCAR code has been developed over years in the frame of a tripartite Institute by the CEA, EDF and Framatome. It is aimed at modelling the behavior of fission products, corrosion products and actinides in the primary circuit. First, we will present the easy-to-use operational abacus elaborated from OSCAR calculations: these abacus enable to assess the amount of fuel released from a disseminating rod, from the iodine 134 specific activity measurements. Second, focusing on an actual case of a French PWR unit, we will show how the code can be used to predict the contamination evolution into the primary circuit, in terms of iodine 134 specific activity, noble gases specific activity and alpha emitter surface activity.

OS5.1 (T5.3-0468)

NextGen RP: Applying Remote and Automated Technologies to Enhance and Optimize Nuclear Power Plant Radiation Protection Operations

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Currently, most radiation measurement and characterization activities that occur at nuclear power plants are conducted manually and on a routine basis regardless of whether conditions warrant the evaluation. Advances in sensor, indoor position systems, and data transmission science and technology have enabled remote and automated operations in many industries. There have been advances in radiation remote monitoring technology (RMT) for non-nuclear power plant purposes (for example, for security purposes) and for environmental monitoring following the Fukushima accident. The combination of remote, automated data transmission/operations technology and advanced radiation monitoring technologies could have the following applications: Plant Area Radiation Monitoring, Worker Radiation Monitoring, Effluent Monitoring, Environmental Monitoring (air, water, and groundwater.) Leveraging advanced technologies to risk-inform and automate radiation protection tasks during operation and emergency situations will not only lead to cost savings and improvements in radiation protection and plant operations but also enhance the health and safety of the workers, the public, and the environment.

Additionally, a review of operations and staffing needs for small modular reactors (SMRs) identified several opportunities for automation of typical radiation protection functions. It was determined that a large fraction of the radiation safety functions at a nuclear power could be streamlined with the implementation of more advanced, remote monitoring technologies, application of advanced data analytics and modeling/trending, and utilization of the radiological information to better inform workers, work processes, and reporting needs.

Ultimately, continuous, real time information of the radiological conditions in power plant areas and the environment will enhance worker and public safety. Data from these measurement devices can be accessed at a central operations center and/or displayed around the plant or on handheld devices (smartphones and tablets) as needed by plant personnel. Advanced data analysis tools and intelligent modeling algorithms can be applied to the continuous data to trend and establish predictions of potential changes to the radiological conditions based on plant events or operational changes. This will allow the site to move towards a condition-based radiation protection paradigm where additional monitoring actions are informed by changes in conditions beyond an established baseline, rather than by a pre-established frequency. Automating the gathering of the data along with the application of advanced data trending and modeling tools will simplify the job function of the radiation protection and As Low As Reasonably Achievable (ALARA) organizations. They can then focus on applying the newfound knowledge of the radiological conditions to optimize worker activities and execute on other aspects of radiation protection.

Technological solutions may exist to address pieces of the issue. However, a platform for integrating the radiological information from multiple devices and advanced data analytics with plant processes and systems is currently not available. EPRI is conducting research to identify, develop, demonstrate, and provide information about technologies and strategies that support efficient and safe risk-informed, condition based, and data driven operations.

Keywords: *Radiation Protection, ALARA, Remote Technologies*

OS5.1 (T5.3-0352)

The Radiation Safety of the 10 MW High Temperature Gas-Cooled Reactor

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The 10 MW High Temperature Gas-Cooled Reactor (HTR-10) is the only pebble-bed type reactor that is currently operating in the world. It adopts helium as primary coolant, nuclear grade graphite as reflectors, and graphite pebbles containing tristructural-isotropic (TRISO) coated particles as fuel element. The very high temperature gas-cooled reactor (VHTR), which is a development of high temperature gas-cooled reactor (HTGR), was identified as one of the six fourth generation advance reactors. Its inherent safety feature, high thermal electricity transformation efficiency, and broad prospects in hydrogen production and high process heat application, have attracted wide public attention, especially after the Fukushima nuclear accident.

In 2011, a measurement of the γ dose rate of typical equipment in the primary circuit and some positions in HTR-10 was performed. It showed that the dust filter in the helium purification system had the maximum of the γ dose rate which was attributed to the trap of the radioactive dust. This maximum value was only 0.58 μSv/h, which was only a little higher than the background (0.14 μSv/h). However, the measurement in 2011 was carried out after a long time from the shutdown of HTR-10 in 2007, which means that only the nuclides of long lifetime can be detected. In 2019, a measurement of the γ dose rate of typical equipment in the primary circuit and some positions in HTR-10 was implemented during a short time after the shutdown of the reactor. In table 1, the γ dose rates at some locations in the primary circuit of HTR-10 from the measurement in the year of 2011 and 2019 were presented. The comparison indicated that the γ dose rate of dust filter in 2019 was much higher than that in 2017. The reason may lie in the detection of the short-lived nuclides as well as the log-lived nuclides in 2019. The current measurement can provide useful information for the maintenance radiation protection and radiation safety evaluation of HTR-10.

Table 1. γ Dose rate at some locations in the primary circuit of HTR-10

Equipment or position	γ dose rate in 2012 (μSv/h)	Equipment or position	γ dose rate in 2019 (μSv/h)
Pipe at the entrance of the HPS	0.24 (5)	Pipe at the entrance of the HPS	0.22 (3)
Lower part of the dust filter	0.58 (12)	Lower part of the dust filter	18.55 (22)
Copper oxide bed	0.13 (3)	Copper oxide bed	0.11 (1)
Piping filter	0.16 (3)	Piping filter	0.66 (3)
Entrance of the HPS room	0.15 (3)	Entrance of the HPS room	0.14 (1)

Keywords: HTR-10, Source term, γ Dose rate

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OS5.1 (T5.3-0078)

Preventive checking of nuclear fuel rods

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For a nuclear reactor to operate safely for its operating staff, it is vital to have leak free fuel rods. To this extent the release of fission products during operation is monitored. However, such methods are corrective and only give information about the whole fuel rod assembly instead of the individual fuel rods. To overcome these drawbacks the NIP-method was developed. Using the NIP-method individual fuel rods are tested for their release of fission products. The NIP-method is based on a method described by Perrotta et al [1] and further developed to reduce cross-contamination and increase accuracy [2].

The NIP itself (Figure 1) consist of a square shaped aluminium tube closed at the bottom and fitted with a lid at the top. After a fuel rod is placed and the lid is toughly closed, the NIP can be flushed with demi-water. After flushing a reference sample is taken and the water is circulated through the NIP. At certain time intervals samples are taken for gamma-spectroscopy. Two regions of interest are monitored: the 662 keV ^{137}Cs and the 1173 keV and 1333 keV ^{60}Co peaks. ^{137}Cs to identify fission product leakage, while ^{60}Co is to determine potential oxidation of the fuel rods.

Identifying leaking fuel rods is more easily compared to non-leaking fuels rods. (i.e. clear peaks in gamma spectrum compared to no peaks against background noise.) Therefore, special care has to be taken in background reduction and subtraction. We would like to illustrate the usefulness of the process by showing our work using the NIP-method at the Hoger Onderwijs Reactor Delft.

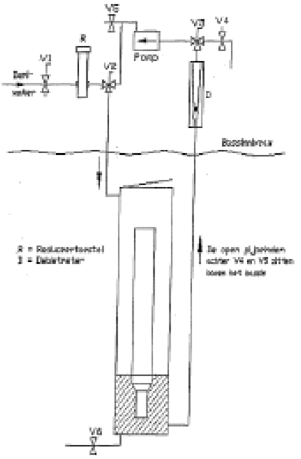

Figure 1: Schematic representation of the NIP with a fuel rod placed. [2]

Keywords: leak testing, fuel rods, NIP

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OS5.2 (T5.4-0074)

INVAP hot cell radiological design, manufacturing and commissioning for radioisotope production plants

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INVAP is an Argentinean company whose main nuclear activities are the design, construction and commissioning of nuclear facilities such as Radioisotope Production Plants, Research Reactors, Fuel Manufacturing Plants, Waste Management Plants, Small Modular Reactors and Nuclear Medical Center Facilities.

In particular, the radioisotope production plants designed by INVAP aim to select, extract, purify and condition fission product radioisotopes for medical and industrial use.

Due to their medical use in diagnosis and treatment, ^{99}Mo and ^{131}I are the most widely produced radioisotopes. They are almost always produced from irradiated fissionable material in a nuclear reactor.

INVAP´s road map to produce radioisotopes is based on using Material Testing Reactors (MTR) with Low Enrichment Uranium (LEU) silicide fuel with dedicated irradiation facilities where LEU aluminum matrix miniplates are irradiated to produce fission products, which are later separated and purified in the radioisotope production plant.

Irradiated LEU plates contain many fission products that emit high energy gammas (with photon energies in the order of 1 MeV) with an activity greater than 6000 Ci (222 TBq).

These high energy gammas shall be shielded in order to ensure low external exposure of the workers. This implies enclosing the radiochemical processes of a radioisotope production plant in hot cells in order to reduce the external exposure of the workers to values in accordance with the latest standards.

INVAP designs for its customers these hot cells which are the main pieces of shieldings of such a plant. This implies performing preliminary and detailed engineering which include mechanical design and shielding calculations.

After the design stage, the hot cells are built at INVAP or a contractor workshop. A hot gamma test is then carried out at the workshop with industrial radioactive sources during the Factory Acceptance Test. The objective of such a test is detecting serious design or construction failures susceptible to cause dose rates so high that the radioisotope production plant could not be operated.

Finally, the hot cells are setup in the radioisotope production plant and a hot gamma test is carried out with the real radioactive sources during the commissioning stage. The objective of such a test is detecting small construction or assembly errors in order to correct them with a shield supplement.

This work presents the different steps from the preliminary design to the final commissioning hot test. It shows deterministic and Monte Carlo simulations of hot cells carried out in INVAP and gives examples of measurement campaigns carried out in past projects.

OS5.2 (T5.4-0475)

Radiation Protection aspects at the High-Luminosity Large Hadron Collider

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To sustain and extend the discovery potential of the Large Hadron Collider (LHC), a major upgrade is planned in 2024-2026. This upgrade will allow increasing the LHC performance in terms of its instantaneous luminosity (rate of collisions) by a factor of five beyond the original design value and the integrated luminosity (total collisions created) by a factor ten. The new configuration, known as High Luminosity LHC (HL-LHC), relies on a number of key innovations that push accelerator technology beyond its present limits. In this context, Radiation Protection (RP) of such complex machine poses challenges not common to any other research laboratory. This work summarize some of the major challenges faced during the design of HL-LHC, from a RP point of view, by discussing the criteria, the approach and the techniques adopted in the evaluation of prompt and residual dose rates, the optimization of interventions accordingly to the ALARA principle, the estimation of the activation of beam line components and the surrounding infrastructure.

***Keywords**: CERN, HL-LHC, Radiation Protection*

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OS5.2 (T5.4-0436)

Compromise of Personnel Protective Clothing from Liquid Exposure

Following critiques of multiple personnel contamination events from entries into the Oak Ridge National Laboratory's Spallation Neutron Source transfer bay, it was considered that the most likely causes for contamination were personnel protective clothing doffing errors or that moisture (sweat) allowed contamination to wick through the protective clothing. Radiological protection staff looked more closely, however, at the specific area of the clothing where contamination was highest; under enhanced lighting and photochromic manipulation, there appeared to have been some type of moisture in the area. Several tests were performed to determine if perspiration had enabled migration of contamination, and to identify what other liquids might have affected contamination transport. It was determined that most common cleaning agents immediately and permanently destroyed the hydrophobic nature of several of the splash-resistant protective clothing materials, allowing for radioactive contamination to penetrate through the material to the worker.

OS5.2 (T5.4-0431)

The radiological source terms in a nuclear fusion experimental facility

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A thermo-nuclear fusion experimental reactor produces a large amount of energy that is mainly transported from the plasma by neutrons and deposited in the machine components, generating nuclear activation. One of the elements used in the fusion reaction is tritium, the radioactive isotope of hydrogen, but in many instances, experimental facilities are based on deuterium only reactions and do not use tritium as fuel. High temperatures in the plasma-facing components and in other parts of these facilities require specific cooling systems that are subject to the neutron activation and that transport the radioactivity into their loops and components.
Because of the above considerations, the radiation sources in an experimental fusion machine could be:

- the primary neutronic field resulting from the fusion reactions occurring in the reaction chamber,
- the gamma radiation generated from neutrons' interaction with the machine components,
- the X and gamma radiation due to the plasma currents,
- the gamma radiation emitted by activated products in the machine components
- loose contamination from activated dust generated in the machine components,
- activated corrosion products generated in the cooling loops after the activation of the inner wall of cooling water pipes,
- activation of the cooling water,
- tritium used as fuel for the fusion reaction or produced in the D-D fusion reactions,
- wastes, still containing tritium and gamma emitters,
- activated air produced in the main hall atmosphere and released to the environment,
- neutrons and secondary gamma radiation generated in Neutral Beam Injectors.

The current analysis is a brief review of the studies on the subject, aimed to define the radiation protection approach to be applied to the next fusion experimental machines. Specific reference will be made to the DTT experimental fusion device which is in an advanced design phase in Italy.
Activities developed at different experimental fusion machines, like TFTR in the USA, JET in England, JT60 in Japan, together with some minor experiments implemented in Italy, will be the basis for identifying the typical radiological source terms. Finally, the studies performed for designing the international project ITER and the Italian DTT will be considered for providing qualitative and quantitative information about the radiological source terms and the potential radioactive waste produced and released to the environment.

Keywords: *Nuclear fusion, Fusion facilities, radiological source terms*

OS5.2 (T5.4-1237)

Design and Performance Prediction of Large-Area Hybrid Gamma Imaging System Using Monte Carlo Simulation

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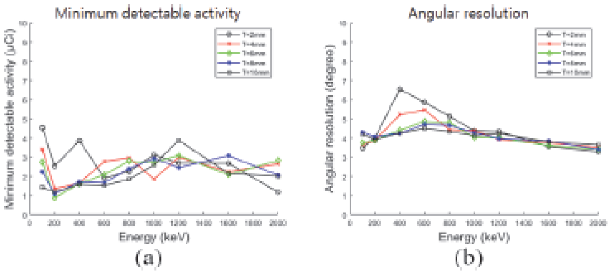
Large-area Compton camera (LACC) [1] was developed for high-speed imaging of radioactive material in large site such as nuclear facilities. While the LACC is specialized in imaging high-energy gamma-rays, a wide dynamic range is desirable since contaminations often contain several isotopes emitting gamma-rays with various energies. A hybrid imaging, combining a mechanical collimation with a Compton imaging, is one of the options to reinforce the performance of the LACC on low energy. In the present study, a large-area hybrid gamma imaging system was designed based on the LACC and its performance was estimated using Geant4 Monte Carlo simulations. The large-area hybrid gamma imaging system is composed of two large-area position-sensitive NaI(Tl) scintillation detectors and a tungsten collimator mask. The detectors inherit their design from the LACC. The mask is based on modified uniformly redundant array pattern, and most of its configuration is adjusted according to the characteristics of the detector. Since the performance of the imaging system is mainly affected by the thickness of the mask, in the present study, the thickness of the mask was determined by Geant4 Monte Carlo simulations. To find a proper mask thickness, minimum detectable activity (MDA) and imaging resolution were studied for several cases with different mask thicknesses (2 – 10 mm) and gamma energies (100 – 2000 keV). For each case, the background radiation, a source at 3 m distance, and the hybrid gamma imaging system were simulated in Geant4. The MDA was derived based on receiver operator characteristic analysis [2], by finding the activity that satisfies criteria of probability of false alarm ($P_f = 0.1$) and probability of detection ($P_d = 0.9$). P_f and P_d were calculated from 250 runs of simulation, for each activity condition (0 – 50 μCi). The MDA and the imaging resolution, estimated for one-minute measurement, are shown in Fig. 1. Performance degradation was observed on thin mask condition. On the other hand, the MDA and the resolution were consistent when the mask had enough thickness. The mask thickness was decided to be 6 mm considering these results. With the decided design, the performance of the system was predicted for a ^{137}Cs source. The imaging resolution was estimated to be 5° using iterative image reconstruction. In the present study, the large-area hybrid gamma imaging system was designed using Monte Carlo simulation. The findings of the present study will be utilized to construct an actual system.



(a) (b)

Fig. 1. MDA (a) and resolution (b) of the system with various source energy and mask thickness conditions

Keywords: *Large-area hybrid gamma imaging system, Minimum detectable activity, Monte Carlo simulation*

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OS5.2 (T5.5-0019)

Holistic approach for radiological risk assessment in the transport of radioactive material in Cuba

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The aim of the study was a holistic approach for radiological risk assessment in the operations related with transportation making for the two carriers of radioactive material by road from Cuba. In the first case, there is the Isotopes Centre (CENTIS), which is designer; consignor and carrier of type A packages with radiopharmaceuticals and labeled compounds. In the other side, the Centre of Protection and Hygiene of Radiations (CPHR), this is the carrier through all the country of radioactive wastes and radioactive sealed sources, in industrial packages (BI-1) and type B packages, respectively.

Two used proactive methods of risk matrix (RM) and the failures and modes effects analysis- FMEA with incident learning system (ILS), have most useful results whose contributing to the improvement of the safety and quality of these practice. A conversion of RM to FMEA, based in standard classification variables for the last and with relationship to workers and public, making a standard causes list and creating of an international incident database, as an integrated method, is the new aspects for risk assessment.

All this applied using the Cuban code SECURE-MR-FMEA version 3.0. The treated medium-level risk with very severe consequences for public included stolen or lost radioactive material and malicious act in spite of their no occurrence in our country. Some obtained results in graphic form as histogram for level-risk, event tree, failure tree and Ishikawa diagram for each process allowed identifying operations with higher importance for risk and the safety measures and causes with most contribution. The shipments with types A or B packages to airport and the cause of no fulfillment of procedures and best practices are the most important contributor. Packages of radioactive material with removable radioactive contamination on the outside of the package or with defective shielding are occurrences with higher frequency and association between model and reported events in database.

This experience may be useful for others in this field towards the radiological safety improvement and a safety culture development with a report culture and the lesson learned.

Keywords: *transport, radioactive material, risk analysis*

OS5.2 (T5.5-0683)

National Network for the Safe Transport of Radioactive Material in the Dominican Republic

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Today's world has undergone significant advances in the implementation of various techniques for social and economic development, among them the different applications and peaceful uses associated with nuclear technology in areas such as: Human health, Agriculture, food security, water resources , environment, energy development, protection and nuclear safety, as well as being of vital importance for the continuous development of the thematic areas mentioned.

The Dominican Republic is not exempt from these applications, use and practices associated with ionizing radiation with the purpose of promoting strategic areas for national development. For all the aforementioned, in adherence to accelerated globalization and commercial exchange throughout the world, there has been a proportional and significant increase in the transport of radioactive material inside and outside our borders; in that sense and by its very nature, the implementation of physical protection and radiological safety measures is justified to mitigate the consequences of potential accidents in order to reduce possible risks to people and the environment, in the case of fissile materials. Criticality risk must be taken into account.

Considering the aforementioned, the Dominican Republic has established a regulation to establish the requirements that must be met to guarantee the safety of the transport of radioactive materials, following the suggestions and requirements of the safety guidelines of the International Atomic Energy Agency (for its acronym in IAEA English). Compliance is mandatory throughout the national territory and applies to all modes of transport, say by land, water or air, as well as all related operations and conditions such as: preparation, dispatch, handling, storage in transit and reception in the final destination of packages containing radioactive material.

In that sense, it has been possible to establish a National Network for the Safe Transport of Radioactive Material where through an inter-institutional agreement it has been possible to integrate all the private and governmental institutions (16 institutions) that are related at some point with transport of radioactive material, thus defining the functions and roles of each of them. The main objective is that at the time of transport all the vectors involved are protected against the direct and indirect effects of ionizing radiation, however, it is not always possible to guarantee the safety of the route for various causes such as weather, traffic problems, obstacles, unforeseen, among others.

Keywords: *Transporte, materiales radiactivos, Red de transporte*

OS5.3 (T5.6-0547)

Derivation of nuclide-specific surface-clearance levels by application of the SUDOQU-methodology

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The current Belgian regulatory framework does not explicitly specify activity levels for the clearance of surface-contaminated objects leaving the controlled area of a nuclear facility. The need for surface-clearance levels was also highlighted in the 2013 IRRS (Integrated Regulatory Review Service) mission conducted by the International Atomic Energy Agency (IAEA), with focus on the Belgian nuclear- and radiation-safety regulatory framework. This led to a collaboration between Bel V and the Dutch National Institute for Public Health and the Environment (RIVM), with the objective of using the SUDOQU methodology [1] for the derivation of nuclide-specific, surface-clearance levels. The SUDOQU methodology, developed by RIVM, allows dose assessments for exposure to surface-contaminated objects. Its innovative feature is the explicit consideration of mass (activity) balance by which surface- and airborne-contamination levels become time dependent and strictly related to each other. This allows for a more realistic evaluation of the dose in public-exposure scenarios, where the exposed subject is more likely to be exposed to a surface-contaminated object for which the activity progressively decreases as a result of its use. An extensive feasibility study on the applicability of SUDOQU for the derivation of surface-clearance levels [2, 3] was conducted through the performance of deterministic calculations over a broad set of scenarios. The results of this study were used as input in the definition of stochastic calculations, more suitable to represent an even wider set of objects and scenarios. An approach for the identification of dose criteria to be used in the determination of the surface-clearance levels was developed. This consists in defining multiple effective-dose criteria, assigned to different percentiles of the effective dose distribution, these criteria not exceeding the annual, public dose limit. Compliance with the annual, local skin-equivalent dose limit for the public is also ensured. Particular attention was devoted to the correct consideration of radioactive progeny and its contribution to the total dose, based on appropriate weighting factors for dose coefficients [4]. The presented results are applicable to clearance of surface-contaminated objects from the controlled area of a nuclear facility, followed by reuse by an adult member of the public. The obtained surface-clearance levels will be shown for approximately 400 radionuclides, and considerations will be made concerning the behavior of the different emitter types and the identification of related trends. Conclusions will be drawn with respect to the practical applicability of the results and their comparison to commonly used, generic surface-activity levels. Further opportunities for development will be highlighted.

Keywords: Clearance, Surface contamination, SUDOQU.

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OS5.3 (T5.4-0303)

Feasibility Study of Compton Camera Computed Tomography (C^3T) by Monte Carlo Simulations

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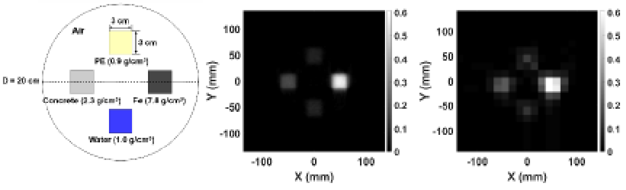
During decommissioning of a nuclear facility, the amount of radioactive waste can be reduced by identifying and removing hot spot-type radiation sources or contaminations in the waste drum. For imaging the hot spots in the waste drum, research is in progress by using Large-area Compton camera (LACC) [1] and a statistical image reconstruction algorithm (ML-EM) that should consider the 3D distribution of linear attenuation coefficients (i.e., attenuation map). The objective of the present study is to demonstrate the feasibility of obtaining the attenuation map using the LACC itself. In the present study, we modeled an LACC-based Compton Camera Computed Tomography (C^3T) system and then predicted the performance of the system by Geant4 Monte Carlo simulations. The LACC-based C^3T was modelled in Geant4, which is composed of a 20 mCi ^{137}Cs gamma-ray beam source, a cylindrical phantom, a drum rotation system (not modelled), and the LACC. In the present study, as a first step of the study, a fan-beam was assumed. The modeled phantom was shown in Fig. 1(a). The phantom was rotated at intervals of 1°, and the imaging time was assumed to be 10 seconds for each projection. The attenuation map was reconstructed by the filtered back projection with Ram-Lak filter, and the reconstructed map was compared with the reconstructed map obtained by an existing industrial gamma CT system. We also estimated the reconstructed error for the reconstructed attenuation map of the C^3T system, and compared the result with that obtained by the existing system. Fig. 1(b) and Fig. 1(c) show the reconstructed attenuation maps obtained by the C^3T and the existing industrial gamma CT system, respectively. As shown in Fig. 1(b), the C^3T system estimates the true linear attenuation coefficients precisely for high-density materials as well as low-density materials. The results show that the attenuation map can be obtained by using the LACC itself. The existing gamma CT system can also reconstruct the attenuation map (Fig. 1(c)); however, the image accuracy was much worse. The reconstructed errors were estimated to be 3.1×10^{-2} and 5.9×10^{-2} for the C^3T and the gamma CT, respectively. The C^3T showed nearly 1.9-times-lower reconstruction error than the conventional gamma CT system. The improvement seems due to the use of monolithic crystal and an advanced interaction position estimation algorithm in the C^3T. The reconstructed attenuation map in the radioactive waste drum will be served as a system matrix in the ML-EM algorithm, providing high-quality images for hot spots in the waste drum.



(a) (b) (c)

Fig. 1. Reconstructed attenuation map using C^3T (b) and industrial gamma CT system (c)

Keywords: *industrial gamma CT, attenuation map, Compton camera*

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OS5.3 (T5.4-0305)

Experimental feasibility study of LACC-based Compton Camera Computed Tomography

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For imaging the location of radiation source or hot-spot contamination in a radioactive waste drum, we are planning to use the Large-area Compton Camera (LACC) [1] with a statistical image reconstruction algorithm (ML-EM). The use of the ML-EM algorithm, however, requires the distribution of linear attenuation coefficient (i.e., attenuation map) for the waste drum. In the present study, we proposed and experimentally demonstrated a new imaging method called Compton Camera Computed Tomography (C^3T), which uses a Compton camera (e.g., LACC) as the detector to obtain the attenuation map. The LACC-based C^3T consists of a ^{137}Cs gamma-ray beam source, a drum rotation system, and the LACC. In the present study, a very low activity (= 250 μCi) ^{137}Cs point source was used as the gamma-ray beam source. The source was located at 2 m distant from the LACC. To test the feasibility of the Compton CT experimentally, a lead brick of 10 cm (L)× 5 cm (W) × 20 cm (H) was placed at the center of the drum rotation system, and the LACC measured the gamma-rays from the ^{137}Cs source while rotating the lead brick using the drum rotation system. The distance between the center of the lead brick and the LACC was 40 cm. The lead brick was rotated through 360° at intervals of 5°. The measurement time was set to very long (= 1 hour per projection) due to the low activity of the source. The attenuation map was then reconstructed by the filtered back projection with Ram-Lak filter. Image reconstruction was done considering a fan beam source. Fig. 1(a) shows the 2D reconstructed attenuation map image for the lead brick with the red square line which indicates the true shape of the brick. The reconstructed image well matches the true shape of the brick. Fig. 1(b) and Fig. 1(c) show 1D profiles for the horizontal (X-axis) and vertical (Y-axis) directions. The results show that the Compton CT can reproduce the value of the true linear attenuation coefficient. The reconstruction error was calculated to be 0.59. Some artifacts were observed in the reconstructed image due to an experimental alignment error during the long measurement time from the low-intensity source. The image quality is expected to be significantly improved by using a higher activity beam source. In the near future, we will carry out various experiments to obtain the attenuation maps for various materials and objects.

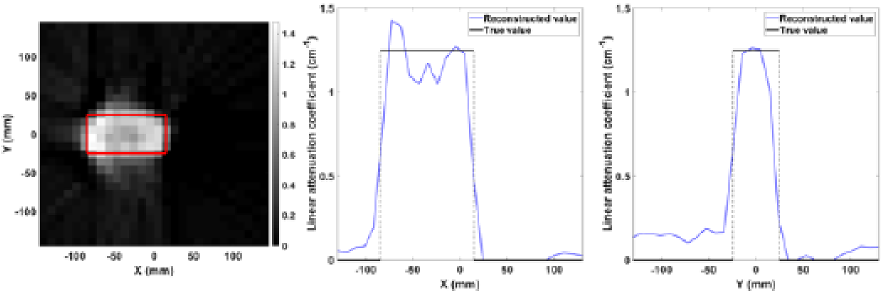

Fig. 1. Reconstructed attenuation map for lead brick (a) and 1-D profile image for X-axis (b) and Y-axis (c)

Keywords: *Compton camera, computed tomography, attenuation map, image reconstruction*

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OS5.3 (T5.6-0248)

Age dependent cancer risk estimations of different radiation protection solutions from a contaminated pond at a nuclear power plant

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Sediments originating from dredging activities in a nuclear site were placed in a pond, which has to be taken into consideration during the future decommissioning process. The sediments have to be handled since a key goal during a decommissioning process of a nuclear power plant is to free release the site where a critical step is to reduce the residual radioactivity in order to meet the site free release criteria. Radionuclide concentrations in several sediment core samples (5 cm diameter and 50 cm length) collected from different locations in the pond were measured and depth profiles were assessed. The radionuclides Co-60 and Cs-137 were identified and the activity concentrations (Bq kg^{-1} wet-weight with reference date in 2014-01-01) were quantified in the range of 10-6000 Bq/kg and 5-50 Bq/kg, respectively. The activity levels were concentrated at the uppermost 10 cm levels. The depth profiles, ranging from 0 to 50 cm, of the analyzed radionuclides were used as input parameters for assessing the radiological impact in terms of gender specific (male and female) lifetime attributable risk (LAR) of cancer incidence attributed to the exposure at various ages. The excess cancer risk predictions were calculated using the raw data of Federal guidance report no. 15 [1] and the risk coefficients from EPA Radiogenic Cancer Risk Models [2]. Three different hypothetical scenarios were used, and an external exposure pathway was considered. First, calculations were performed for direct external exposure from the sediment. Thereafter calculations were made where layers of the sediments were removed. Calculations were also performed where layers of soil were added on the sediments. The results in Fig.1 shows the different risk outcome to a male member of the public exposed for a year for the three different scenarios mentioned above. As an example, for a 30-years-old man exposed during one year without any adjustment of the soil and when the pond is emptied on water the excess cancer risk was predicted to be 0.00255, which recalculated to effective dose corresponds to 7.4 mSv/year. E.g. adding 5 cm of soil on the surface will accomplish a dose reduction of 57 %. As nuclear power plants are decommissioned there is a need to perform cost-benefit analyses regarding radiological risks and different management solutions for areas with residual activity.

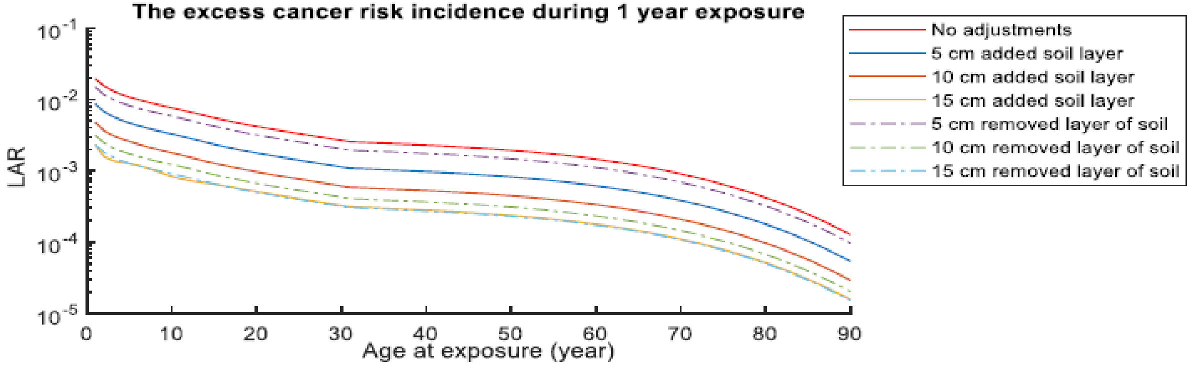

Fig. 1. The increased cancer risk for exposure at various ages (0 to 90 years) of 1-yr fulltime exposure from the pond without water when different amount of soil is added or removed respectively.

Keywords: Decontamination, external exposure, general public

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OS5.4 (T5.7-0105)

NORM waste management: impact on residents surrounding a landfill

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French regulators have taken advantage of the transposition of the European directive 2013/59/Euratom to update the regulation on the management of waste from industries involving NORM. The activity concentrations of waste containing NORM are highly variable depending on the raw materials used in the industrial processes and on the processes themselves, from a few Bq/kg to several thousand Bq/kg of uranium 238, thorium 232 and/or their daughters, as well as potassium 40.

One of the evolutions is that NORM waste whose activity concentrations is lower than 20,000 Bq/kg of uranium 238 and/or thorium 232 may be stored in a landfill for usual dangerous substances, and not necessarily in a radioactive waste storage facility.

The impact on residents living in the vicinity of such landfill may be assessed on the basis of various scenarios. These scenarios must cover the operating phase of the landfill and the post-closure phase. For each phase, both normal operations and accidental situations should be assessed.
The aim of this presentation is to describe these scenarios and the associated hypotheses:

- the type of waste (pasty or powdery, rich in uranium 238 or rich in thorium 232 or in potassium 40),
- the exposure parameters to calculate external and internal exposure (distances between the radiation source and the residents, suspension factors, inhalation rates, exposure times, food consumption rates, etc.).

On the basis of the results that will be presented, some recommendations may be made to the authorities and to the operator of the landfill for the protection of the public).

***Keywords**: NORM, waste management, public members*

OS5.4 (T5.7-0516)

Improved radiological characterization of large and heavy magnets for radioactive waste elimination

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The operation of particle accelerators at CERN requires the use of electromagnets, which are accelerator components needed to guide and focus charged particles. In the framework of maintenance activities and machine upgrades or dismantling, a large number of magnets have been removed from the accelerator complex. The scope of the paper is to present an improved characterization validation method of Very Low Level Waste (VLLW, TFA for "Très Faible Activité" in French) magnets [1, 2]. Systematic calculations with the analytical code ActiWiz were performed to study the activation scenarios which can lead to the production of radioactive nuclides in magnets from the CERN proton machines, in the frame of their possible elimination towards the final repositories managed by ANDRA (Agence National pour la gestion des Déchets RAdioactifs). This study allowed the establishment of a radionuclide inventory required by ANDRA. Such inventory corresponds to the list of radionuclides which can be produced with activity levels above the declaration limits furnished by the repositories. The fingerprints for the different types of magnets are provided as a function of the decay time. The evaluation of absolute levels of specific activity can be obtained by normalizing the average dose-rate in contact with a magnet with global conversion factors, giving a process to characterize CERN's electromagnets without recurring to any gamma spectroscopy measurement. In order to validate the methodology, we developed a framework to calculate and reduce activity uncertainties that results from the gamma spectroscopy assay technique. These uncertainties originate from the geometry model description used to compute efficiency calibration curves within ISOCS [3]. The objective is to identify the best geometry models that fit multiple spectroscopy counts or activity lines. We show in this paper that the transfer function method is reasonably penalizing (hence slightly overestimating the activities) compared to the gamma spectroscopy results. The agreement with the gamma spectroscopy results indicate that transfer functions provide a valid method for calculating the referred three-radionuclide activities (Fig 1. And Fig 2.). It also indicates that the transfer functions activity results are consistently conservative.

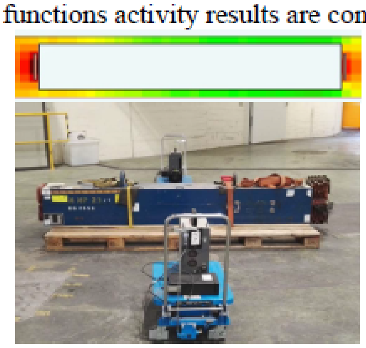

Fig. 1. Top: Dose rate mapping. Bottom: Gamma spectroscopy measurement setup.

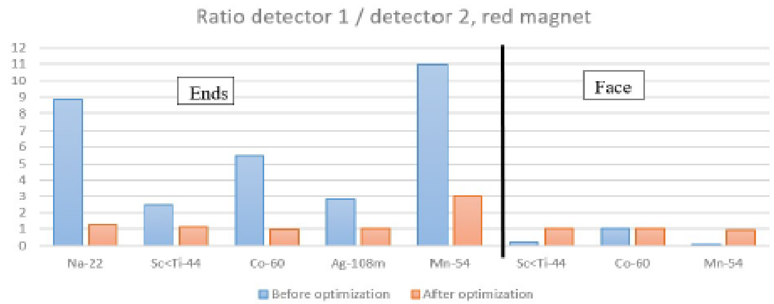

Fig. 2. Ratio between opposite scans activities at the ends and faces of a magnet. A ratio close to 1 shows a satisfactory activity optimization.

Keywords: Radioactive Waste characterization, gamma spectroscopy, Non-destructive assay techniques

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OS5.4 (T5.7-0562)

Planning of the Retrieval of Intermediate Level Waste from the Asse II Mine

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The Asse II is a former potassium salt and rock salt mine starting operation in the beginning of the 20th century. The operation was ended in 1964, when the German Federal Ministry of Research took over the mine and started emplacement of radioactive waste in 1967 until 1978. During this time 125.787 drums with low level waste (LLW) and 1.301 drums with intermediate level waste (ILW) were emplaced in different chambers in the salt mine Asse II.

Due to the fact, that a long-term safety could not be reached the German parliament decided in 2013 that all the radioactive waste has to be retrieved out of the Asse mine. DMT and several subcontractors performed the conceptual planning by order of the operator BGE to retrieve the ILW from the Asse mine stored only in emplacement chamber 8a/511.

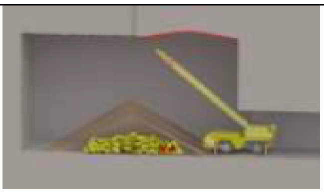
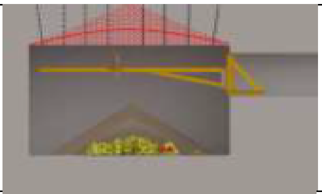
A first study delivered the following results:

a) In general, all activities for retrieval have to be carried out remotely controlled or automatically.

b) Although in 1996 measurements have shown dose rates above 1 Sv/h close to the drums, an MCNP simulation delivered dose rates in the range of 1 mSv/h close to the entrance of the chamber. That leads to the possibility that personnel can get a very restricted access e.g. in case of intervention. As some drums are no longer tight and radioactive aerosols are already existent in the aerosol filter, personnel entering the chamber for a short time have to wear respiratory protection.

c) Additional exploration results especially for the mining stability of the chamber are needed. These results are essential for the decision, which technical way of retrieval has to be used.

Depending on different situation of the stability of the chamber DMT planned 3 different ways for retrieval:

1. A basic concept, where the floor can be used by special vehicles and the ridge can be stabilized for example by rock bolts. Drums will be lifted and put into specific boxes. The boxes will be inserted into container without cross-contamination and brought to the shaft, where the container is lifted to the surface.	
2. An alternative concept, where the floor cannot be used by vehicles a crane is lifting the drums from the top and put them into specific boxes.	
3. In case ridge and floor have not enough stability, a specific partial back filling of the chamber can be executed to stabilize the chamber before drilling a new entrance to the chamber for retrieval.	

The following paper will show pros and cons of the different alternative ways, how the ways can be used for the retrieval of the drums and how radiation protection requirements will be kept.

OS5.4 (T5.7-0140)

ON PERMANENT GEOLOGIC REPOSITORY FOR SPENT NUCLEAR FUEL: WHAT CAN UAE CARBONATES OFFER?

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Spent Nuclear Fuel is "fuel that has been withdrawn from a nuclear reactor following irradiation, the constituent elements of which have not been separated by reprocessing".
In the common case of "once-through" nuclear fuel cycle (that is, no reprocessing of waste), the spent fuel is usually stored on site in spent fuel pools of water to cool down for about 5 years, and thereafter transferred to dry cask for interim storage that could extend for a period of several decades, with the intention of quenching the heat generation and lowering the toxicity.
The extension of the interim storage period to 100 years or even more has been suggested, in the hope of developing better technology through R&D efforts for final disposal.
Geologic repositories are the only permanent spent-fuel disposal options considered today.
Beyond developing a regulatory framework of procedures and guidelines for an SNF repository project, States set up an appropriate National infrastructural, including:

- Setting-up an "implementing organization" to carry out SNF repository site selection activities,
- Setting-up an efficient financial system for the "implementing organization",
- Defining the "Stakeholders" and their relationships with the "implementing organization".

The "Implementer" uses site-suitability criteria to reduce prospective settings to potential sites to candidate sites to a single proposed site. Knowledge acquired from scientific and technical investigations carried out at the surface, as well as in underground research laboratories, are used in the technical filtering process.
Any permanent repository system of spent nuclear fuel relies on its ability to impede the diffusion of radionuclides outside the system so that the requirements specified by national regulations are fulfilled. To that effect, the disposal concepts used in the design of such repository systems for spent nuclear fuel use a multiple, independent and often redundant barriers; both natural and man-made (engineered).
In a diversified geology, the use of "specific site-suitability criteria" to filter-out sites from day one is believed inappropriate and technically questionable, as the data needed for such an approach are not available at the early stage of siting. US DOE called for the use of "Generic Criteria" to be used to provide the initial foundation for site-suitability criteria, despite their limitations. The DOE's 1984 Siting Guidelines are an example of Generic Criteria. However, DOE's 2001 Yucca Mountain-specific site-suitability regulation, relies on probabilistic performance assessment.
In addition, public opinion has become so important that governments had to use two filters in site selection: Technical Suitability Filter and Societal Acceptability Filter. That led to the fact that long-term disposal plans of SNF have been derailed in several countries. Until now, the selected geologic formations are:

- Salt (Germany, USA),
- Crystalline Rock (Finland (committed), Sweden (committed), France, Canada, China, Japan, UK, USA),
- Clay/Shale (Canada, China, France (committed), Japan, UK, USA),
- Volcanic Tuff (Yucca Mountain, USA)

Currently, only Finland, Sweden and France are committed to geological disposal of spent nuclear fuel, and they all opted for mined geological repositories. While France opted Clay as a host rock, Crystalline rocks have been selected as host rocks in Sweden and Finland. The two concepts envision a repository system at shallow (500m) depths, composed of both natural and engineered barriers, to be constructed using conventional mining techniques.
Deep Borehole Disposal has been studied recently by experts and international organizations. In this case, The SNF waste is proposed to be emplaced in deep boreholes (3-5 Km), where the host-rock environment is relatively stagnant, density-stratified, hydro-geologic system, and events like earthquakes, or human intrusion, would be much less likely to disturb the waste. Waste canisters containing the spent fuel would be emplaced in the lower 2 Km of the borehole. Researchers and experts from different countries and organizations have called for further R&D efforts to study the concept and investigate issues related to drilling technology, geo-mechanics, geophysics, geo-chemistry, hydrology, hydrogeology, canister design and retrievability, ….

This paper presents "*Generic Criteria*" to provide the initial foundation for site-suitability criteria pertinent to the carbonate geologic formations of the UAE and their physical characteristics *vis-à-vis* permanent geological disposal of spent nuclear fuel.
Carbonates have complex pore size distributions, leading to wide permeability variations for the same total porosity, making it difficult to predict their fluid flow, or sealing, characteristics.
Quantitative characterization of carbonates is a difficult challenge. They are:

- Highly Heterogeneous (many lithofacies; hydraulically connected through capillary pressure: same Pressure but different Saturation at boundary,
- Highly Anisotropic,

OS5.4 (T5.7-0140)

ON PERMANENT GEOLOGIC REPOSITORY FOR SPENT NUCLEAR FUEL: WHAT CAN UAE CARBONATES OFFER?

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- Tensorial stress/strain fields,
- Highly faulted (seismic & sub-seismic) & fractured (all sub-seismic),
- Low compressibility (Cv),
- High variation in density,
- High variation in thermal conductivity (Kth) (mineralogy),
- High variation in sound wave propagation velocity (tensorial) & travel time (Vp, Vs, Rho),
- Karstic formation,
- Vuggy (super permeability).

Advanced techniques, like model-based interpretation methodology and artificial neural networks for rock type analyses, are used to evaluate carbonates.

Most of the world's giant fields (like in the Arabian Gulf region) produce hydrocarbons from carbonate reservoirs.

Correspondingly, there are two principal (cap rock) sealing formations in the UAE:

- Nahr Umr shale (sealing the Thamama Group), and
- Hith Anhydrite (sealing the underlying Arab Formation),

Secondary seals and barriers also exist in the stratigraphic sequence.

The cap rocks are relatively impermeable rock, commonly shale, anhydrite or salt, that form a barrier or seal above and around the reservoirs underneath so that fluids cannot migrate beyond the reservoir. It is often found atop a salt dome. The permeability of a cap rock capable of retaining fluids through geologic time is ~ 10-6 to10-8 Darcies.

Nahr Umr: Nahr Umr is a shale formation found throughout the southern part of the Arabian Gulf and forms the cap rock to many major reservoirs in the region. In the UAE, Nahr Umr Formation is a series of shales, siltstones and mudstones, with increasing carbonate content toward the north of UAE. The percentage of clay minerals in the shale ranges from 62% to 73%, while the non-clay fraction is dominated by calcite, quartz and traces of dolomite, feldspar pyrite, glauconite and phosphate. Nahr Umr thickness ranges from 63m to 220m and forms an excellent cap rock to more than 5,000 reservoirs underneath. Geochemical studies indicate that the shales have no oil or gas generating potential; the total organic carbon content ranges from a low of 0.20% to a high of 0.50%. These sediments were deposited in a shallow marine to distal offshore shelf setting.

However, recent literature reports major wellbore instability problems in the UAE when drilling through Nahr Umr Shale Formation. Well-bore instability were noticed in new wells and in re-entry wells, especially with the rise of water-based muds and stricter environmental control, making wellbore stability in this shale an extremely challenging operation for drilling/mud engineers. Hence, Nahr Umr might not be suitable as a host rock for SNF repository. Further investigation of Nahr Umr should include coring with non-intrusive (neutral) drilling muds to acquire scientific and technical knowledge (Rock Typing, Porosity, Permeability, Capillarity, Faults (sealed, open), Fractures (sealed, open), Stylolites, Karstic conduits, tensorial Stress & Strain Fields, Density, Compressibility (horizontal vs vertical), Thermal conductivity,…..).

Hith Formation: There is only little published description of the Hith Formation beyond a cursory description of the nodular and chicken-wire character of the anhydrite and its thickness. It is believed that the depositional settings of the Hith Formation in the subsurface match those of the present-day supratidal evaporite depositional setting along the Arabian Gulf coastal area. However, in contrast, the Hith represents a mega-environment covering a much greater geographical area and produced a much thicker sequence of sediments dominated by evaporites, with much less carbonate than in the Holocene sabkha sediments of Abu Dhabi. The ultimate top seal for this sequence in western Abu Dhabi was previously thought to be the Hith Anhydrite, but more recent literature suggests that it might rather be the tight limestones of the basal Cretaceous Habshan Formation. Evidence for this is that sour gas of similar composition to that in the Arab-ABC, has been found in the Manifa Formation that overlies the Hith Anhydrite in the same area. Oil and gas with high H2S content has also been found in Habshan. This suggests that sour gas has migrated into even higher overlying reservoirs, possibly along fault planes. The lack of data on Hith calls for in-depth characterization of Hith to ascertain its sealing capability.

OS6.1 (T6.2-0218)

Dose Assessment in Population at the Remote Period after the Chernobyl Accident

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Traditionally, in order to assess the internal doses, two types of models are used: deterministic and phenomenological. First type is based on the determination of transfer factors by trophic chain: soil → plant → animal → human. Deterministic models include ecological models of different types, such as ECOSYS-87, which includes all the pathways of radionuclide entries. The disadvantage of these models is that the diversity of soil types causes variation of transfer factors from soil to plants up to hundreds times.
The second type of model is phenomenological, that is based on experimentally obtained data characterizing radiation situation. Radionuclides intake with food is represented by two components: products of animal origin, equivalent to the daily consumption of 1 liter of milk, and other components of the diet and drinking water, equivalent to the daily consumption of 1.5 kg of potatoes. The main disadvantages of phenomenological model are: model is not able to adequately react on changes of the diet, in particular, the increase in consumption of forest food products; model estimates are several times higher than the actual dose values, calculated according to the WB-measurements.
These models disadvantages increase the level of conservatism in the dose evaluation. Therefore, the development of a new method for internal dose evaluation was essential.
It would make sense to use the WBC data. Internal dose assessment based on WB-measurements is more accurate and reliable as it was caused by ^{137}Cs intake with a real diet.
In the late 80s a system of dose monitoring of the residents living in contaminated with Chernobyl radionuclides territories was established. There are 35 WBC. Database of the WBC measurements contains over 3 million records.
In addition to direct factors in the modeling should be included indirect factors, which influence the dose formation. The smaller the settlement, the farther it is from the local center, the worse are the social and economic conditions of life of its residents, and the greater is the degree of naturalization on private farms. Consumption of contaminated foods and internal dose in small settlements is higher than in large ones. Data analysis of WB-measurements confirmed this. Numerous studies, including ours, revealed exclusively important role of the availability of forest foods for rural residents in the internal dose formation. Transfer factor in the chain soil-milk is influenced internal dose. Farmland soil type determines transfer factor.
According to the ICRP recommendations, the remote period after the accident refers to a new concept of 'existing exposure'. Under these conditions, with the requirements of the ICRP the concept of a representative person, as a representative member of the most exposed critical group among inhabitants of a settlement, was introduced.
To maintain continuity and conservatism in the dose assessment for radiation safety of the population living in contaminated territories, evaluations were conducted for the representative person in the most exposed group of residents of a settlement.
All rural settlements have been classified by regional types of soil, causing the intake of ^{137}Cs with foods, population size and forest access. There are three classes. Regression model of the internal dose of the representative person, calculated according to the WB-measurements, on density of soil contamination has been constructed for each region. For this purpose the entire range of density of soil contamination of settlements in each region was divided into 6 intervals. Model prediction error was 30 - 45%, which shows its high-quality.
According to the developed method, a Catalogue of Average Annual Effective Doses of Residents of Belarus has been established. The Catalogue of Average Annual Effective Doses of Residents of Belarus, along with the density of soil contamination of the settlements is the basis for next in turn regular document of the Council of Ministers of the Republic of Belarus for classification of the settlements by zones of radioactive contamination. It will be actual for the period from 2021 to 2025.

OS6.1 (T6.2-0625)

Lessons learned from Chernobyl: Assessment of the radiological consequences and formation of potential radiological risk groups

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The article describes the major results of long-term radiation epidemiological studies of health effects after the Chernobyl accident performed in the Russian National Radiation-Epidemiological Registry (NRER) [1] during 30 years. The main purposes of the studies were identification of the groups of potential radiation risk of death from cancer and circulatory system diseases (CSD) among Russian Chernobyl liquidators [Fig 1]. The groups of potential risk were identified by values of excess relative risk (ERR) and relative risk (RR) estimated in the cohort of 134 thousand people for the period of follow up from 1986 through 2017, the average radiation dose is 0.11 Gy, maximum individual dose is about 1 Gy. Values of excess relative risk of solid cancer incidence and mortality per 1 Gy (ERR Gy-1) are 0.47 (95% CI: 0.03; 0.96, p-value=0.034) and 0.58 (95% CI: 0.002; 1.25, p-value=0.049), respectively. The radiation excess solid cancer cases were estimated as 233. The percentages attributable to radiation were 3.9%, 5.2%, 7.5%, 9.1% and 13% at dose ranges of 50-100, 100-150, 150-200, 200-250, and 250 mGy and higher, respectively. During the follow-up period 1986-2014, 157 cases of leukemia were detected (with the exception of chronic lymphocytic leukemia). During only the first period 1986–1997 was established a statistically significant ($p<0.05$) linear dose dependence of the leukemia incidence with an excess relative risk of ERR/Gy=4.17 (90% CI: 0.18, 13.24. For the period of follow up from 1986-2012, 12400 deaths were caused by CSD. The group of potential radiation risk comprises of the liquidators with accumulated doses 0.15 Gy and above, they arrived in the Chernobyl exclusion zone in the first year after the accident and stayed there less than 6 weeks. The total size of the identified group was 9.5 thousand people (7% of the cohort members). In the group of potential radiation risk the statistically significant value of RR is 1.44 (95% CI: 1.25; 1.66). In this risk group 31% of deaths from CSD should be considered as associated with radiation. In the beginning of 2013 the size of the group of potential radiation risk was 6155 liquidators. For remained life about 950 radiation induced deaths from CSD are expected.

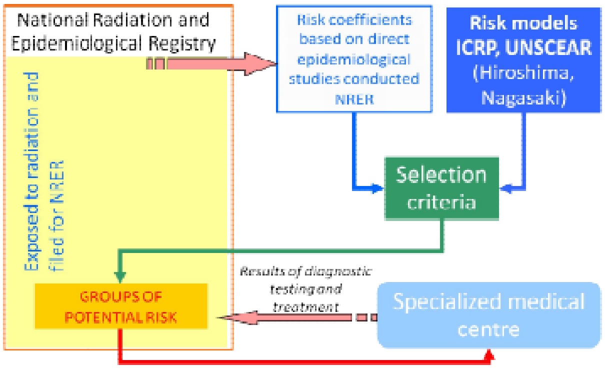

Fig. 1. Identification of potential risk groups

Keywords: Radiation risk, Cancer, Circulatory system diseases, Chernobyl accident.

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OS6.1 (T6.2-0209)

Current Status of the Environmental Monitoring Database on the Accident at Fukushima Daiichi Nuclear Power Plant

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Since the nuclear accident at the Fukushima Daiichi Nuclear Power Plant, many organizations conducted environmental monitoring to understand the distribution of radionuclides and take countermeasures. However, the results of environmental monitoring were prepared in different formats according to the purpose of each organization and were published on their websites respectively. Even if the data was collected and the format was unified, it was difficult for ordinary people to understand the distribution of radiation doses and for experts to find the target data. Furthermore, the data for the eight years after the accident became enormous, and it became more difficult to understand the entire data by merely converting the data into maps and charts. In order to solve these problems, the Japan Atomic Energy Agency (JAEA) developed and improved the Environmental Monitoring Database that makes it easy to use these measurement data (Fig. 1).

In this database, measurement data could be downloaded and used in CSV, KML, and XML formats. In particular, for XML formats, JAEA defined a standard format "EMML" that considered quality confirmation and long-term use for various environmental monitoring data. Besides, maps and charts were created from these measurement data so that they could be displayed on the web site and downloaded as image data. These maps and charts were created with some unified styles and legends so that the data measured at different times, locations, and methods could be compared with each other. In addition, a large amount of data for eight years after the accident was statistically analyzed to make it easy to confirm the entire data. Box-and-whisker plots and histograms of measurement data were prepared so that variations and transitions in radiation levels from year to year could be confirmed. As a tool to assist users of the database, we also developed a visualization tool that allows users to create maps and charts in specific areas that they want to display. The visualization tool can access the database directly and use the latest measurement data. Users can also create maps and charts from data that is not in the database by preparing the data in CSV format.

Currently, the database has registered and published more than 200 types of air dose rate and radioactive concentration data (over 800 million records). With the release of these measurement data, the database recorded over 20,000 pages of access per month and 50 GB of data download per month. There were also accesses to databases from more than 80 countries. Environmental monitoring is still necessary not only for the areas under evacuation orders but also for the areas where the residents live to confirm their safety. These measurement data will be collected and published, and the database will be improved according to the users' needs at that time.

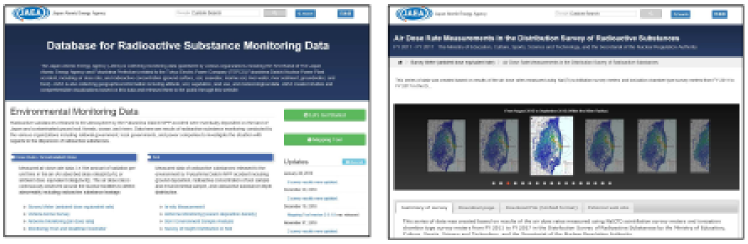

Fig. 1. The Enviromental Monitoring Database (left: top page, right: summary page)

Keywords: Fukushima Daiichi Nuclear Power Plant, environmental monitoring, database

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OS6.1 (T6.2-0153)

The Joint Environmental Radiation Survey around the Fukushima Daiichi Nuclear Power Plant between JAEA and KAERI

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According to the implementing arrangement between JAEA and KAERI in the field of the radiation protection and environmental radiation monitoring, the joint environmental radiation survey was conducted to assess the radioactive cesium deposition in the ground around the Fukushima Daiichi nuclear power plants (FDNPP) on October, 2018. The technical purpose of environmental radiation survey was to make a field application of developed survey system and method, because their performances in the contaminated area were very important things with respect to the appropriate response to the nuclear accident. Therefore, all survey results and methods between two institutes were shared and compared to be experimentally verified during the joint survey. As shown in Fig. 1, four kinds of gamma-ray spectrometers were used in the ground-based and mobile gamma-ray spectrometry, which were two potable HPGe detectors by the institute, MS_ERS (multipurpose system for environment radiation survey) based on $LaBr_3(Ce)$ detector [1] of KAERI, and KURAMA (Kyoto University Radiation Mapping) system with CsI(Tl) detector [2] of JAEA. First, the ground-based gamma-ray spectrometry using portable HPGe detectors and MS_ERS was performed in 6 sites around the FDNPP to assess the depth profile and deposition density of radioactive cesium in the ground. In addition, the ambient dose equivalent rate as well as dose rate of ^{134}Cs and ^{137}Cs were determined from a measured energy spectrum at a fixed position in the site. The mobile gamma-ray spectrometry using backpack and carborne survey platforms was conducted to evaluate ambient dose equivalent rate around the survey sites and the carborne survey route from site to site, respectively.

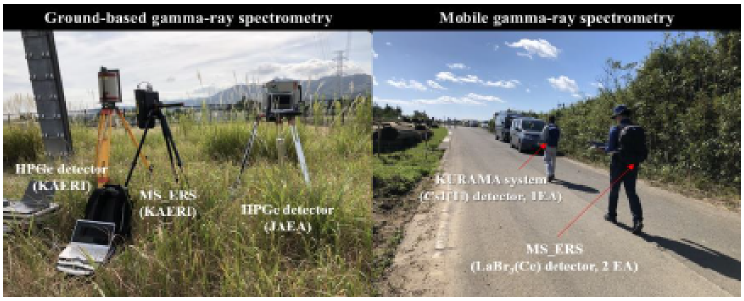

Fig. 1. The joint environmental radiation survey around the FDNPP

Keywords: Environmental radiation survey, radioactive cesium, gamma-ray spectrometry

ACKNOWLEDGMENTS
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OS6.1 (T6.2-0152)

Estimation of External Gamma Doses from Deposited Radionuclide on Inhabited Areas of Korea

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Studies for understanding gamma external dose from the radionuclide deposited on urban inhabited area have been conducted after the Fukushima nuclear accident in Japan. Several months after the Fukushima accident, it was needed to evaluate dose reduction factors of the buildings in inhabited areas for determining a criterion value for the designation of the intensive contamination survey area. It was known that radiocesium is the major nuclide contributing to ambient dose rates in the area affected by the Fukushima accident. Several experts have studied the distribution of radiocesium inventories on components in urban areas in the evacuation zone affected by the accident to provide the essential parameters for dose evaluation models and to better understand the migration of radiocesium deposited in urban areas.

Fukushima accident has showed that the information on the dose reduction factors of the buildings in inhabited area is essential for the selection of the long-term countermeasure. Japan had difficulties in conducting the long-term emergency preparedness due to the lack of the data on the dose reduction factors of the buildings in inhabited area after Fukushima accident.

In Korea, the early phase emergency preparedness system against a nuclear accident has established. But this system is not appropriate for supporting long term emergency preparedness. One of the essential data for establishing site specific long term emergency preparedness system is the dose reduction factors of the buildings in inhabited area. It is because that the biggest contribution on the radiation effect in long term phase is the external gamma dose given from the deposited radionuclide on inhabited area.

In this study, a series of air kerma calculations at various locations were performed for the 9 cases of the domestic buildings in inhabited areas and for three radiation source energies of 0.3 MeV, 0.662 MeV and 3.0 MeV; and therefore, the dose reduction factors were derived with the air kerma data set. These results would contribute to be used as one of reference materials in establishing emergency preparedness planning against a nuclear accident.

Keywords: *Long term phase, Inhabited area, Air kerma*

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OS6.1 (T2.5-0501)

Environmental monitoring for emergency preparedness: 30 years of experience at ENVINET

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Environmental monitoring, radiation protection and emergency preparedness are closely linked topics. Radiation monitoring networks for the environment are installed either as nation-wide networks or at sites that are considered as possible sources for unintended radioactive releases. Depending on their location and purpose, networks are constructed in location-specific ways using adapted components. Most networks are composed of fixed installations, supplemented by a number of mobile devices.

Low-power GDR (Gamma Dose Rate) probes like MIRA can endure week-long power outages and still reliably provide authorities with vital dose rate information. Spectroscopic probes like SARA offer nuclide specific information in real-time to help detail response plans depending on the expected biological effects in case of accidents. They also render the identification of unexpected sources possible. Especially for such cases, supplemental aerosol monitors like SIRA allow to detect smallest traces of radionuclides in the air and thereby considerably extend response time for unreported events. Highly sensitive, mobile spectroscopic detectors like MONA are used to prepare detailed, nuclide-specific radiological pre- and post-event base-maps to assess evacuation and decontamination needs. Especially, they provide close-to-real-time maps of nuclide distributions during emergencies, either car- or airborne.

Ideally, all data is collected in a central monitoring software like NMC. Such a software supports field-personnel as well as local operators and decision makers by mission-specific data preparation techniques and presentation layers. Data exchange functionalities need to be available for immediate data sharing between all authorities and entities involved in CBRN. They also allow for a coupling to dispersion calculations systems to enhance nuclide dispersion prediction calculations, release origin identification as well as agency-specific decision-making systems.

During the last 30 years, the contribution of environmental monitoring system to radiological preparedness has been continuously changing. In recent years, a notable trend is visible towards spectroscopic solutions, even for nation-wide monitoring networks, either as main monitoring source or superimposed on a dense ultra-low-power GDR-only network. Ring monitoring networks are upgraded to allow nuclide identification for dose rates up to 100 mSv/h. Gamma-spectroscopic aerosol monitors start to displace classic α/β-samplers in nation-wide networks. Other notable developments include the use of drones and tailored airborne solutions. In this talk, views and experience of a commercial probe manufacturer regarding some of these developments will be presented.

OS6.2 (T6.1-0552)

The Role of the IAEA Safety Standards in the Area of Emergency Preparedness and Response

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The IAEA's Statute authorizes the IAEA to establish or adopt — in consultation and, where appropriate, in collaboration with the competent organs of the United Nations and the specialized agencies concerned — standards of safety for the protection of health and the minimization of danger to life and property, and to provide for the application of these standards. The IAEA commenced its safety standards programme in 1958.

The first Safety Standard in the area of emergency preparedness and response (EPR), Safety Requirement No. GS-R-2 (Preparedness and Response for a Nuclear or Radiological Emergency) was issued in 2002 and co-sponsored by seven international organizations. This Safety Requirements publication established the requirements for an adequate level of preparedness and response for a nuclear or radiological emergency in any State. In 2015, it was superseded by Safety Requirement No. GSR Part 7 (Preparedness and Response for a Nuclear or Radiological Emergency), which was co-sponsored by 13 international organizations (FAO, IAEA, ICAO, ILO, IMO, INTERPOL, OECD/NEA, PAHO, CTBTO, UNEP, OCHA, WHO and WMO). Currently, the set of Safety Standards in the area of EPR includes Safety Requirement No. GSR Part 7 and three Safety Guides, while two additional Safety Guides are at the final stage of preparation or publication.

The safety standards in the area of EPR are based on the scientific considerations derived from the findings of the United Nations Scientific Committee on the Effects of Atomic Radiation (UNSCEAR). They take into account the recommendations of international expert bodies, notably the International Commission on Radiological Protection (ICRP), and provide for their practical application.

One of the examples of a link between the fundamental science of UNSCEAR, the recommendations of the ICRP and the requirements of the IAEA safety standards in EPR is related to the protection strategy for the public in case of a nuclear or radiological emergency. GSR Part 7 requires that "protection strategies are developed, justified and optimized at the preparedness stage for taking protective actions and other response actions effectively in a nuclear or radiological emergency to achieve the goals of emergency response." Safety Requirements explain the importance of the application of different criteria for different purposes. Specifically, they explain the application of reference levels that are applicable to the overall protection strategy; the generic criteria for individual protective actions; and the triggers (such as conditions on the scene, operational intervention levels and emergency action levels) for initiating the different parts of an emergency response plan. They outline how to start developing this strategy from setting up a reference level from ICRP recommendations to establishing a set of generic criteria and to developing default triggers.

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Generic criteria are based on the current knowledge of deterministic and stochastic effects, as outlined by UNSCEAR. They address both external and internal exposure that could be directly related to the full range of important radionuclides. There is also a plain language explanation associated with each of the criteria.

Keywords: *Emergency, Protection, Criteria*

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OS6.2 (T6.3-0555)

Integration and modernization of country size early warning systems to improve the nuclear emergency preparedness of Hungary

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Radiological, nuclear and chemical facilities are potential threads for people working or living in these areas. The key points in accidence management are reliable information about the situation and quick reaction time for responders.

The factories usually build their own monitoring system to know what is the situation in there technology. The information coming from this source is not always suitable for the need of first responders, because the low level of integrity and /or the long transmission time of the data. The disaster management of Hungary decided to build their own independent system. The project was launched in 2006 and will be finished in 2022. The data coming from chemical, radiological and meteorological sensors, more than 1000 pcs, are arriving directly to redundant data centrals. The communication channel used by the system will be available in case of emergency situations (earthquake, power out, etc....) as well. The system, with the help of redundancy and automatic repair function, remains operational despite several defective components. multiply components malfunction. The system provides information about the situation of the dangerous factories and the surrounding environment, and to send automatic warning to responders, and to provide immediate interaction with people affected by the danger. Special tools (online siren network, local displays, and website) are available to broadcast information directly to the public.

First responders are carrying sensors during their mission, these data can be available for remote analysis, and the mission control can be centralized and supported with tools like spread calculation algorithm. The system is able to cooperate with mobile laboratories and radiation shielded vehicles.

The system is includes the data coming from independent sources, like the technology monitoring system of factories, and neighbor countries exchange services. A necessary part of this system is the data export function to send actual data to international organization and partner countries.

In this project monitoring stations were installed on the territory of Slovakia and direct data exchange was developed in the framework of bilateral agreements.

This presentation will focus on the process of designing of such a system and sharing many years of operational experience and further development possibilities like improving international data exchange standards of radiological data.

OS6.2 (T6.1-0138)

Australia's international review missions and implications for emergency preparedness and response

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Australia has undergone two international peer review missions less than twelve months apart, each including aspects relating to emergency preparedness and response to a nuclear or radiological accident. These missions included all nine Australian jurisdictions, and may be summarized as follows:

- **Joint External Evaluation** (JEE, facilitated by World Health Organization (WHO), December 2017) of the implementation of the *International Health Regulations (2005)* (IHR). The aim is the performance of a comprehensive review of Australia's health security, including the ability to respond to radiation-based emergencies.
- **Integrated Regulatory Review Service** (IRRS, facilitated by International Atomic Energy Agency (IAEA), November 2018). This IRRS mission was a peer review of Australia's regulatory frameworks for nuclear and radiation safety, and included a module on Emergency Preparedness and Response (EPR).

National action plans for both reviews have been developed, and those relating to EPR for nuclear/radiological events will be presented. One of the most significant challenges identified for Australia is the establishment of a national framework for radiation safety that assures a consistent level of safety and protection of people and the environment across all jurisdictions.

A key deliverable for the implementation of international best practice in EPR across all jurisdictions of Australia is the recently published *Guide for Radiation Protection in Emergency Exposure Situations (2019)*. Provided in two volumes, the Guide is a tool for implementing the framework requirements of the IAEA's *General Safety Requirements (GSR) Part 7, Preparedness and Response for a Nuclear or Radiological Emergency*. Additionally, practical advice is provided on planning, preparedness, response and transition phases of the emergency (see Figure 1), as well as information on the protection of emergency workers and the use of the hazard assessment to inform a graded approach to protection.

This oral presentation will include a description of various outcomes from the reviews, the current status of the action plans and a summary of the above-mentioned guidance. Future plans will also be discussed.

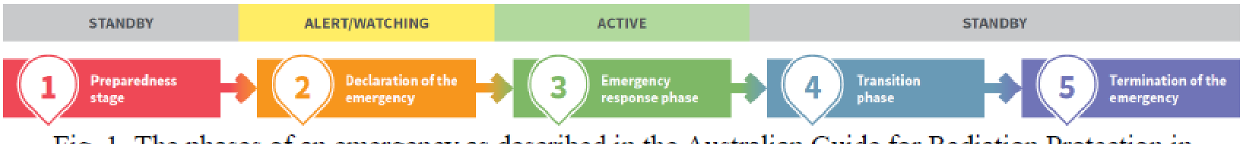

Fig. 1. The phases of an emergency as described in the Australian Guide for Radiation Protection in Emergency Exposure Situations (2019)

Keywords: *Peer Review, EPR, Health*

OS6.2 (T6.2-0529)

Integrating mental health and psychosocial support in radiological or nuclear emergency planning, response and recovery

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Health impairment due to a nuclear or radiological accident always includes impairments attributable to both ionising radiation and non-radiological effects, such as psycho-social and societal impairments. As seen from Chernobyl and Fukushima experiences, the latter had a significant socio-economic impact on the affected communities. It is unclear how these aspects could be included in a quantitative fashion in considerations at the preparedness stage and in decision- making during an emergency.
Within this context, the OECD Nuclear Energy Agency Working party on Nuclear Emergency Matters (WPNEM) has decided to set a dedicated Expert Group (named EGNR for Expert Group on Non Radiological public health aspects) to examine how to consider psycho-socio and societal impact of protection actions, and develop mitigation of these aspects.
Currently used protective strategies for planning for, response to and recovery from radiological or nuclear emergencies need to be revisited and enriched by applying a holistic societal approach taking into consideration psycho-social and mental health impact of the emergencies themselves and of the response and recovery actions. Launched in November 2019, the EGNR will develop practical tools/solutions for mitigation of that aspect and to assist emergency management decision makers. This work will build upon the WHO guidance on a policy framework for Mental Health and PsychoSocial Support (MHPSS) in radiological and nuclear emergencies under preparation (1), and a dedicated international workshop co-organised by BfS, OECD-NEA and WHO in March 2020 (2). The findings of this workshop will help the EGNR to propose generic recommendations that can be adapted to national and local issues and concerns according to the prevailing circumstances. A "whole-of-society" approach (*i.e.* promoting inclusiveness of all stakeholder categories) and an "all-hazards" approach will guarantee that the decision-making process will come out with the best protective actions in a multifaceted fashion integrating in itself cascading events of an evolving emergency cycle. This presentation will discuss the main findings of the workshop and the further development of a practical tools/check lists for MHPSS by the EGNR in 2020-21.

Keywords: Psychosocial support, emergency preparedness, practical tools

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OS6.2 (T6.5-0526)

Building a Framework for Preparedness for Post-Accident Recovery Management

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The area of recovery management (RM) has been of interest within the OECD Nuclear Energy Agency (NEA)'s Committee on Radiological Protection and Public Health (CRPPH) for some time, in particular since the 2011 accident at the Fukushima Daiichi nuclear power plant. An aspect of RM that has been identified as needing further practical guidance is advanced planning for post-accident recovery actions. Within this framework, the CRPPH Expert Group on Recovery Management (EGRM) was created at the beginning of 2019 with the main objective of assisting NEA member countries in planning and improving their preparedness for recovery by producing guidance on how to develop a nuclear or radiological post-accident recovery management framework which can be adapted to national conditions. The final aim is to prepare a short, comprehensive and operational generic framework of recovery management easily adaptable by any member country before any accident happens. This framework will address a series of issues at stake for post-accident management, ranging from food management, drinking water management, urban and environmental decontamination, waste management, balance of radiological and psycho-social effects of decisions, stakeholder involvement to communications processes. In addition to the content of this framework, the guidance will propose a process for involving relevant stakeholder in the preparedness phase with collaborative deliberation on the issues at stake. As an accident situation moves from the emergency phase to the transition phase and on to the recovery phase, the shift of roles and responsibilities will also be addressed. During the recovery phase, governmental management aspects become more focused on support to affected stakeholders rather than decisional, which can be extremely resource intensive. There is thus a need at the planning stage to be prepared to "create" resources and skills to address these different challenges. This presentation will discuss planning approaches to address these needs.

Keywords: *Preparedness for recovery, nuclear or radiological accident*

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OS6.2 (T6.2-0440)

Balancing Pros and Cons of Emergency Countermeasures

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Both during and after a nuclear emergency, decision makers are faced with complicated dilemma's: do the health benefits of evacuation outweigh the anxiety that it may cause the public? Up to what point are the costs for decontaminating a city justified, and will people still be willing to live there afterwards? Tools are available to assess some of the separate aspects of these dilemma's: for example, several models exists to assess if the monetary costs of a measure outweigh its benefits, and lessons have been learnt from the accident at Fukushima about the health consequences of evacuating the elderly and hospital patients. Tools to combine different aspects of measures - radiological, psycho-social and financial, as well as acceptation by the public and practicability, however, still are lacking, so that in an emergency decision makers would be left to compare apples and oranges.

We will present the results from a study that aims, eventually, to lead to the development of tools that enable decision makers to balance the diverse pros and cons of emergency measures.

OS6.3 (T6.4-0084)

Application of cellular technologies in the treatment of local radiation injuries

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One of the most common types of radiation damage exposed to ionizing radiation appear local radiation injuries (LRI). LRIs are very difficult to treat, conservative treatment of non-healing chronic radiation ulcers are ineffective, and surgical intervention is often required, which is not always possible due to the condition of the body.

One of the promising methods of treatment of LRI is cell therapy by mesenchymal stem cells (MSC), as well as their paracrine factors. So it is shown that the use of bone marrow MSC improves the course of skin LRI (decrease fibrosis, improved collagen organization) and accelerates wound healing processes. Today, an alternative source of MSC is gingival mucous tissue. They are identical to bone marrow cells, readily available, have high proliferative and regenerative potential, but there are no studies on their use in the treatment of LRI.

The purpose of this work was to study the processes of regeneration in the treatment of LRI of the skin of MSCs of the mucous membrane of human gums and their conditioned medium in laboratory animals.

In our work, the MSC mucous tissue of the human gums and their conditioned medium were used, the test system was Wistar rats, X-ray lnk installation 268. We carried out a planimetric analysis, histological and immunohistochemical research and statistical data processing. In the experiment, non-personalized MSC samples were used, which are on long-term cryo-storage in a biobank. Thawed MSCs (seeded in a vial at passage 3) were cultivated up to passage 5.

OS6.3 (T6.4-0084)

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When 80-90% of confluence was reached, the cells were removed and used for administration to animals, and the conditioned medium for concentration (CCM) was also collected. To obtain CCM, a LabScale tangential filtration system was developed for laboratory use, designed for concentration, diafiltration and microfiltration. So made the concentration of conditioned medium 10 times. The resulting volume was collected in syringes for subsequent administration to animals. The calculation of the dose of the introduction of conditioned medium was performed taking into account the concentration of cells.

80 male Wistar rats weighing, on average, 200 grams randomized into 4 groups depending on the type of therapy produced.

1st group - control (C), in which animals did not receive therapy;

2nd group - control with intradermal administration of the culture medium concentrate (CM)

3rd group - intradermal injection of MSC of the mucous membrane of the human gums

4th group - intradermal administration of CCM from MSC of human mucous gums

Therapy was performed at 1, 14, 21 days of the experiment.

The animals were injected into anesthesia with the help of Zoletil, they shaved their hair, fastened it to a foam substrate and simulated skin LRI at a dose of 110 G on the LNK-268 unit with specified parameters.

OS6.3 (T6.4-0084)

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When the planimetric assessment of the modified skin of animals on 7th day in the CCM group, the area was significantly larger compared to the other groups C, MSC, CM ($p \leq 0.05$). On the 14th day, all animals showed the appearance of wet dermatitis, as well as an increase in the size of the affected skin area with the formation of an open wound surface.

On the 28th day, the appearance of exfoliated necrotized epidermis covering the wound surface of the skin was registered in all animals. In the CM group, the area of the ulcer was significantly less compared to all 3 groups of C, MSC and CCM. Until the 42nd day of the study, the dynamics of a decrease in the area of the ulcer skin of animals was observed in all groups. The area of skin ulcers was less in the CM group when compared with C, MSC and CCM groups.

And in group C, from 42th to 77th days of observation, an increase in the area of the skin ulcer was noted compared with the CM and CCM groups. From 91th days to the end of the observation, there were no statistically significant differences in the parameters of the total area of the altered skin and the area of the ulcer in all groups. On day 112, complete healing of the ulcer was observed in the CM group in 40%, in the MSC group in 60%, and in the CCM group, in only 20% of the animals; in group C there was not a single animal with a prolonged wound defect.

OS6.3 (T6.4-0084)

Application of cellular technologies in the treatment of local radiation injuries

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In all groups during the experiment purulent-hemorrhagic skin lesions, inflammation of the dermis with signs of infiltration and fibrosis were observed in samples of excised tissue. In group C, all days of the study in animals observed a marginal thickening of the epidermis, sometimes with symptoms of acanthosis. Subcutaneous muscle during the whole period of observation was with signs of infiltration. Up to 70th days in the CM group, the area of the epithelized surface did not change, and then in some cases relapses were observed (reduction in the area of epithelialization and an increase in purulent-hemorrhagic surfaces).

Hair follicles and their rudiments (1-3 follicles in sight) began to appear actively from the 56th day. In the subcutaneous fat and subcutaneous muscle, the phenomena of edema and infiltration (up to 70th days) were followed by fibrosis (from 91th days). In the MSC group, processes of lymphocytic infiltration were observed up to the 56th day of the derma to be, and from 70th day there was a decrease in these processes and fibrosis of the dermis and focal proliferation of connective tissue in SFT and subcutaneous muscle. An increase in the number of hair follicle buds from 56th day to the end of observation.

OS6.3 (T6.4-0084)

Application of cellular technologies in the treatment of local radiation injuries

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An increase in the number of microvascular vessels was noted from 70th day until the end of observation. To the CCM group on the 70th day, the ulcers were completely epithelized, and on the 91th and 112th days, the formation of ulcerated surfaces was observed again. The processes of marginal epithelialization were identified on 28th day and persisted throughout the study. Subject derma on all days of observation was with signs of fibrosis, lymphocytic infiltration, as well as granulation and necrosis. Hair follicles were found on 91th and 112th days in 66% of the animals.

Also produced immunohistochemical analysis of the markers:

Marker newly formed vessels CD31, CD68 macrophage marker, endothelial growth factor VEGF, nerve fiber regeneration marker and hair follicles PGP 9.5, a marker of inflammation Ki67, mediator exudative-destructive inflammation fibrinogen F8, matrix metalloproteinases (MMP) 2 and 9 type, fibroblast growth and keratinocytes TIMP2 and collagens 1 and 3 types.

These markers were evaluated in absolute amount in samples of excised tissue. Significant differences within groups between 112th and 28th days are highlighted in red. We can note that in the group of MSC and CCM the number of newly formed vessels, nerve fibers and hair follicles mount, the number of mediators of exudative inflammation increases.

OS6.3 (T6.4-0084)

Application of cellular technologies in the treatment of local radiation injuries

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In groups C and CM in place with an increase in the number of nerve CL16 fibers, markers of inflammatory processes increase, especially in the group C.

The activity of the following markers was evaluated semi-quantitatively in points from 0 to 3, where 0 is a complete lack of expression, and 3 is pronounced expression

The activity of these markers in varying degrees by the end of the experiment varies depending on the healing of the burn injury.

So, all the used treatment methods, including 3-fold administration of MSC, culture and conditional medium concentrates at a dose of 2 million per 1 kg were effective in skin LRI and resulted in a reduction in the area of damage, accelerated healing of the ulcer, and improvement of the regenerative processes. In addition, the use of mesenchymal stem cells of the mucous membrane of the human gums led to an improvement in vascularization and a decrease in inflammatory processes in the source of radiation injury.

OS6.3 (T6.2-0626)

On-site Radiation Emergency Medical Preparedness of KHNP

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In Korea, National Radiation Emergency Medical System (NREMS) provides medical response to nuclear power plant accident. In addition, Korea Hydro & Nuclear Power Company (KHNP) has been operating its own medical response organization. When disaster occurs in nuclear power plant (NPP), Radiation Health Institute (RHI) affiliated to KHNP since 1996, located in Healthcare Innovation Park of Seoul National University Bundang Hospital, dispatch radiation emergency medical preparedness team (NPP dispatch team) which is consist of personnel such as medical doctor, nurse, health physics and supports, with mobile dosimetry laboratory, advanced ambulance and decontamination system. The other five teams including off-site medical, situation, dosimetry, administrative and assistance team provide advanced medical care to the NPP workers, response the radiation medical issues and risk communication. From the lessons of the Fukushima nuclear accident, the need for organization to take immediate medical response at the NPP site emerged. In December 2014, the Radiation Emergency Medical Center (REMC) was launched at each NPP site. Three nurses, one emergency aid worker and one support are on 24-hour duty. It was equipped with advanced ambulance and emergency medical facility. In case of radiation emergency, on-site radiation medical clinic is organized by NPP dispatch team and REMC at REMC facility of NPP site, controlled by emergency operation facility (EOF) and provide on-site prehospital care and transportation to hospital in cooperation with NREMS. Recently, as safety issues are highlighted by earthquakes in Korea, on-site emergency facility with isolation system which is headquarter facility of NPP disaster that played important role in the Fukushima nuclear accident are planned to be built in all of the NPP sites. The place isolates facility from ground to absorb the energy of earthquake. It will have medical facility including emergency room, dosimetry room, admission room, isolation room and EOF. When on-site facilities collapse, situationally changeable off-site emergency response preparedness, located at the borderline between controlled area and safe area, is considered using existing EOF buildings, schools or public buildings. It functions as basecamp of NPP's radiation emergency response including medical response, as like J-village in Fukushima. In the future, further considerations of disasters other than earthquakes, multi-unit NPP accident and road breakdown are needed.

Fig.1. Radiation Emergency Medical Center

Keywords: *Nuclear power plants, Radiation, Emergency*

OS6.3 (T6.4-0220)

Current Russian Approaches to the Use of Iodine Thyroid Blocking as a Protective Measure in a Radiation Accident

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In case of radiation accident, which has high probability of the release of radioactive iodine, iodine thyroid blocking is necessary. That is one of the protective measures that apply to humans. Iodine thyroid blocking using stable iodine applies to reduce the effects of radioactive iodine on the thyroid gland. It is proved that the preliminary use of stable iodine shortly before, or simultaneously with influence of radioactive iodine, can effectively prevent the absorption of radioactive iodine by the thyroid gland. Iodine thyroid blocking prevents the formation of higher radiation doses in children and adolescents, as the result the positive effect for these population groups will be higher. If taking of stable iodine is delay, their effectiveness will be rapidly decrease over time, which is require more careful attitude to the application of this protective measure. In areas where a natural deficiency of iodine is attend, levels of radioactive iodine isotopes accumulation in the thyroid gland can be higher, which is able to lead to a higher risk of radiation-induced thyroid cancer.

At the present time, in Russia, in addition to situational criteria, remain in force and apply dose criteria for iodine thyroid blocking at two levels: 250 mGy - 2500 mGy for adults and 100 mGy - 1000 mGy for children [1] In this case, the decision to carry out of iodine thyroid blocking is set on the basis of comparing the predicted dose with the preventable dose for the first 10 days. The presence of dose levels in the Russian regulatory framework makes it possible to use derived intervention levels for managerial decision-making [2] and for iodine thyroid blocking carrying out, primarily for individual sensitive groups (Table 1.) of the population. Since 2010, the "Guidelines for iodine prophylaxis in the event of a radiation accident" [3] has been in force, this document is largely based on the practice of iodine thyroid blocking during the Chernobyl accident and presents Russian approaches and basic principles of iodine thyroid blocking, including brief information on biological dangers of radioactive isotopes of iodine, information on the required dosages and contraindications when prescribing potassium iodide to different population groups. Issues that require discussion without which the proper implementation of iodine thyroid blocking is impossible and which are also presented in the Russian approaches are zoning of the territory for the purpose of organizing iodine thyroid blocking, the distribution of stable iodine, the order of administration of stable iodine, and the creation of reserves of stable iodine.

Table 1.Dose levels of planning for iodine thyroid blocking

Population group	Expected Thyroid Dose Levels, mGy
Newborns, Children and Adolescents	50
Pregnant women	50
Breastfeeding mother	250
Adults up to 45 years old	250
Adults over 45	2500

Keywords: Iodine Thyroid Blocking, Protective Measure, Radiation Accident

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OS6.3 (T6.2-0513)

Management Experience in Acute Radiation Syndrome in the Russian Federation from 1949 to 2018

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For almost a 70-year period, at least 77 radiation accidents occurred in the territory of the former USSR (1949-1991) and the Russian Federation (1992-2018), accompanied by irradiation of people diagnosed with acute radiation sickness (ARS) in 356 victims, including cases burdened by local radiation injuries. In total, as a result of radiation exposure in the first 3-4 months, 68 people died with a diagnosis of severe and extremely severe ARS. The most serious clinical consequences were observed after the accident at the Chernobyl nuclear power plant in 1986, as a result of which 30 people died (two directly as a result of a reactor explosion and 28 as a result of radiation exposure). The diagnosis of ARS was made to 134 victims. By grade of severity there were: 21 - very severe (grade IV), 22 - severe (grade III), 50 - moderate (grade II) and 41 - mild (grade I). In 54 cases, the clinical course of ARS was aggravated by severe skin radiation burns.

The majority of the most seriously injured in the considered radiation accidents were treated in the clinic of the State Research Center — Burnazyan Federal Medical Biophysical Center of FMBA. Computerized medical histories of the victims are in the "Database on acute radiation injuries of people", which has been maintained since 1985. Brief information with individual data on victims of radiation accidents with a clinical picture of the development of the acute period of acute radiation sickness is published in English in the monograph[1]. The information contained in the database is a unique information resource (about 2/3 of the world experience) for studying the clinic of human radiation pathology and the methods of treatment used, which can be useful to support decision-making by medical specialists in the diagnosis and treatment of acute radiation injuries.

Over a 70-year period, as a result of experimental and clinical studies, the pathogenesis and various forms of ARS have been studied in detail, a treatment regimen has been created and adopted, and highly effective means of preventing and treating this pathology have been developed and implemented. Modern scientifically based treatment of ARS at the present stage includes aseptic regimen, means of stimulating hematopoiesis (in particular, growth factors) and intensive treatment of multiple organ failure. The treatment of intestinal syndrome in ARS is improving. Criteria and terms for necrectomy, timely prosthetics and plastic surgery, the use of mesenchymal stem cells in local radiation injuries are determined.

***Keywords**: Radiation accidents, Acute radiation syndrome, Chernobyl accident*

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OS6.3 (T6.4-0593)

Classification of Local Radiation Injuries by ICD and implications for the medical management

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A Local Radiation Injury (LRI) is one caused by the exposure to high doses of ionizing radiation over a short period of time, impacting the skin and underlying tissues. Depending mainly on the locally absorbed doses, severe medical consequences to the affected tissues can be expected.

Due to its particular pathophysiology and inflammatory changes [1], the main clinical characteristic of LRI is the dynamic clinical presentation, which develops in time, sometimes in an unpredictable manner of successive inflammatory waves occurring weeks and even years after radiation exposure [2]. These aspects are part of the complex diagnosis, management and medical follow-up of the LRI [3].

In the medical area is common to find the term "radiation burn" to refer to a LRI. The International Classification of Diseases (ICD-11) for Mortality and Morbidity Statistics defines a "burn" as an injury to the tissues caused by a pathological flux of energy that causes cellular destruction and irreversible denaturation of proteins and is primarily caused by thermal or other acute trauma. Effects caused by ionizing radiation are explicitly excluded from this definition [4].

The current ICD-11 does not provide an accurate codification accounting for the characteristics of LRI as a particular disease entity. Although some codes in ICD-11 provide certain approximation, as the ICD-11 code EJ70 "Acute effects of ionizing radiation on the skin"[4], is incomplete and does not comply with the whole spectrum of LRI. Most of the codes do not consider damage to underlying tissues, neither the multiple recurrences observed in many of the affected patients during their evolution [5], independently from the etiology of the radiation exposure (accidental or not)[6].

The implications are relevant not only for the medical community and affected patients, but also for health providers and administrators. For example, financial support for the management of severe LRI in specialized centers in the last two decades has been extremely difficult to obtain and has jeopardized the proper handling of the condition. An appropriate register of LRI would contribute to an improvement in notification, registering, treatment and medical follow-up. Even though several aspects on this regard have been improved in the ICD-11, an adequate classification regarding the LRI as a particular clinical entity is still missing and should be included on the next ICD.

Keywords: *Diagnosis, Classification, Local Radiation Injury*

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OS6.4 (T6.D-0122)

How experts and affected population can work together in post-accident situations

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The feedback experience from Chernobyl and Fukushima nuclear accidents show the complexity of post-accident situations. In France and more generally in Europe, several reflections have been conducted in order to be better prepared (e.g. French CODIRPA and European research programs on post-accident issues such as PREPARE, SHAMISEN, CONFIDENCE, TERRITORIES). This work highlighted the need for radiological protection experts to engage themselves with the affected population in working together in order to provide appropriate protection and rehabilitation of decent living conditions. However, this dimension is not yet clearly integrated into the existing action plans. This is why the Institute for Radiological Protection and Nuclear Safety, the French institutional body for nuclear research and expertise, has undertaken a reflection on how experts can be at the service of the populations residing in the territories affected by a nuclear accident. This work, still ongoing, has already identified three components: ensuring the long-term presence of experts on the spot; building the capacity of people to make informed decisions; relaying the local concerns to the central level of the Institute involved in the management of the crisis at the national level. More specifically, the possible missions of the experts locally could be to:
- Promote the development of a practical radiological protection culture;
- Support the implementation of a co-expertise process (experts working with the population and not for the population);
- Engage a dialogue with the population about the radiological risk;
- Support access to measurements (environment, products, individuals) including self-measurements (citizen science);
- Develop awareness and capability of local intermediate people/professionals.

***Keywords**: post-accident management; co-expertise; practical radiological protection culture*

OS6.4 (T6.B-0171)

Public Perception of Nuclear Radiation: An educational dialogue with undergraduate students

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Radiation education remains the fodder for back-burner debate, given the strong emotional attachment involved. Since the Fukushima accident, the Division of Human Health (NAHU), International Atomic Energy Agency (IAEA), along with Hiroshima University, Nagasaki University, Fukushima Medical University and National University of Singapore, has emphasized the importance of incorporating aspects of science-technology-society (STS) into medical education. Radiation and Society – is a seminar-style module at the Tembusu College, National University of Singapore, consisted of weekly 3-hour interactive discussion sessions and it is run for 13 weeks in the semester. Undergraduate students from different faculties were exposed to themes and concepts related to radiation and society. Student numbers were restricted to 15 enrollments per group to facilitate effective interactions. It explored the topics including the fear of radiation, historical dimension, complexities of nuclear radiation disaster response, radiation science, risk communication and medical aspects of radiation. A variety of topics – including nuclear fall-out, historical contextualization of nuclear fear/stigma, most importantly, the uses of radiation in biomedicine, STS and science communication. Two field visits were arranged – one to a research reactor and the other one to radiation oncology department at the hospital – in order to make students understand the peaceful uses of nuclear radiation in a variety of fields. Expert guests were invited to share their perspectives on matters including technological developments and societal impacts of radiation.

The facilitator - student interactive discussions during the course of 13 weeks has helped in educating the young minds on the subject of nuclear radiation. A survey was conducted to obtain the perception of enrolled students on the reliability and safety of nuclear energy, the effectiveness of the seminar/course and their opinion on where the different energy sources stand on different fronts. Student opinions were collected both at the start of the semester and after completion of 13 weeks of seminars. Overall findings of the survey showed that although nuclear energy was perceived to offer better safety and reliability, renewable energy was perceived to be the better option. Students also felt that the weekly sessions were effective in the stated objective of providing education on nuclear energy.

The author has successfully conducted this course as a facilitator since 2015 at the Tembusu College, National University of Singapore and more than 100 students complete the course in the last 4 years. This seminar-style module is expected to equip the students with the analytical tools required to assess and question the sources of knowledge as well as social perceptions of radiation. Such educational compunction efforts will groom future citizens to be aware of the pros and cons of nuclear radiation and to respond to nuclear and radiological emergencies.

Keywords*: Nuclear Radiation, Science-Technology-Society, Radiation Education*

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Oral Session T6

OS6.4 (T6.B-0238)

Strategic planning in communication to the public in risk management and management of nuclear emergency in CNAAA

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This paper aims to establish a strategic planning for public communication in risk management and nuclear emergency management at the Almirante Álvaro Alberto Nuclear Power Plant, CNAAA, since the absence of continuous planning increases the vulnerability of communities affected by the lack of permanent training. In an emergency at the Angra I or II nuclear plants, it may cause lack of communication and information to stakeholders, which is divided into various interest cycles and information needs produced to meet their specificities in those cycles, always cautiously and judiciously disseminated to the public parties seeking to meet the interests of the parties [1]. Planning makes it possible to efficiently build communication channels with the public, as well as to provide authorities with strategies to disseminate these communications between cycles, to produce release models and disclosures in the risk management phase and to create performance indices to evaluate these [2]. Strategies in relation to the stakeholders in the communication process, because only with a strategic planning implemented we will create a continuous improvement in these information feedback processes with the increase of the communications quality.

Keywords*: Public Communication, Risk Management, Nuclear Emergency.*

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OS6.4 (T6.B-0246)

The perception of actual and potential nuclear accidents in France (1986-2018)

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For over 30 years the French Institute for Radiation Protection and Nuclear Safety (IRSN) has been following the evolution of risk perception in France, with a particular focus on the risks associated with nuclear power. A public opinion survey is conducted every year among a sample of 1,000 people representative of the French population. It includes a core of questions which are kept from one year to the other and allow the study of the perception of different risks and their evolution over time. As such, it is a valuable tool to analyze risk perception by the French population, its evolution, and to design better risk management policies.

The perception of potential accidents has been a subject of study very early on in the Barometer, but especially after the Chernobyl accident in 1986. This issue came again under scrutiny after the Fukushima Daiichi accident in 2011. When questioned about the catastrophic potential of different types of industrial activities, the French population strongly singles out nuclear power plants, before chemical plants, radioactive waste disposals and the transport of hazardous substances. It underlines the fact that when people envision a nuclear accident, they tend to picture a large scale event. In the same vein, they respond almost unanimously (95 % of them in 2017) that "were an accident to happen in France, it could have dire consequences". It also explains why "the risk of an accident" is the first item people mention as a strong argument against nuclear energy (35% in 2018), far ahead of radioactive waste (23 %), with a remarkable stability over the past 20 years.

Focusing more specifically on the accidents of Chernobyl and Fukushima Daiichi, the Barometer tells us they affect only marginally the opinion of French people on the safety of their own NPPs. One perception they do affect is the perception of an accident's possibility in France, which decreases as time passes. One fact remains, however: the Chernobyl and Fukushima nuclear accidents are the events which have frightened the most the French population, far ahead of events such as Hurricane Katrina (5%) or Haiti's 2013 earthquake (9%). An interesting evolution of late is that the Chernobyl accident is now mentioned as more frightening by the French people than the Fukushima accident, and this was before HBO's Chernobyl revived the tragic memory.

Keywords*: perception, nuclear, accident*

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OS6.4 (T6.B-0408)

When used in communication to general public, does plume maps lead to desired protective actions?

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A Finnish Radiation risk perception survey was made in collaboration with STUK and FMI. Survey responses (1124 respondents in total) represent the Finnish adult society by age, gender based on quotas by region. Data collection was done in 18.12.2017-9.1.2018.

Objectives for the survey were to gather information about attitudes towards radiation, investigate misconceptions and gaps in knowledge about radiation and maps as a communication tool during a hypothetical radiological accident. Special attention was given for the question "How general public perceive the need of protective actions when plume maps are used as a communication tool".

In Finland, communication during radiological accidents requires a strong collaboration between the responsible authorities. Required maps are produced in collaboration with The Finnish meteorological institute and The Finnish Radiation Authority STUK. Survey results are used in decisions making about the maps and other communication material produced for Finnish general public in the future.
Survey results also answer to following questions, to support preparedness for radiological accidents from communications' point of view.

- Does peoples level of knowledge predict behavior during hypothetical radiological accident?
- What kind of actions might be taken and what kind of behavior would occur when different kinds of maps are used in communication to the general public
- What are likely factors that lead to spontaneous evacuation?
- Does parents behave differently during hypothetical radiological accident?

***Keywords**: Preparedness, visuals, plume maps*

OS6.4 (T6.B-0465)

Nuclear and Radiological Emergency Preparedness: the role of universities, research bodies and professional organizations

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An adequate preparedness for radiological and nuclear emergency response, including the post-accident recovery, requires continuous improvement efforts, with the implication of all relevant stakeholders. Public administration bodies (at national, regional and local levels) as well as all public and private civil groups with a role to play should be engaged. An obstacle is the fragmentation of competences between central, regional and local administration levels. During the emergency phase, to avoid deficient coordination, a single chain of command is necessary. However, for the development and running of the intermediate and recovery phases, decentralization of management and local administration levels are essential, with the involvement of the different stakeholders.

Through exercises and practical work with panels of stakeholders, several European research projects, developed in connection with the European Platform on preparedness for nuclear and radiological emergency response and recovery (NERIS), have demonstrated the importance of their involvement. Efforts should be made to increase the radiological protection culture of the different actors, stakeholders and the general population. Apart from technical matters, the key points for improvement also depend on their participation, motivation and engagement. Performing periodical exercises and analyzing the respective roles for realistic accident scenarios usually have a positive impact. Decision support systems can be used to facilitate identifying the main challenges, functions to fulfil and management alternatives, in the search for the best options [1]. These joint exercises help to create a network between experts and different stakeholder groups that in case of a nuclear or radiological emergency should work together although usually are not used to do. Their cooperation will contribute to a better knowledge between them and will ease a more efficient work in case of real situation of emergency.

This is a big challenge and to contribute to its success the participation of “neutral” actors as universities, research teams or professional associations (like the IRPA Associate Societies) is a very important factor. Based on their expertise, they can play a role in assessing and communicating the radiological impacts after accidents and the effectiveness of countermeasures. Their facilitating action could be very important in emergency preparedness to improve the development and implementation of appropriate protection strategies as part of the consequence management and the transition to recovery. They can also help to create a more adequate and transparent communication between the different stakeholders’ groups, to improve better mutual understanding.

Keywords: *Emergency preparedness, stakeholder engagement, post-accident recovery*

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OS6.5 (T6.2-0077)

THE CONFIDENCE PROJECT

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On the first of January 2017, the CONFIDENCE (COping with uNcertainties For Improved modelling and DEcision making in Nuclear emergenCiEs) project started as Task 9.1 of the European H2020 project CONCERT. CONFIDENCE brought together more than 30 partners from all over Europe to perform research on identifying and reducing uncertainties in the release and post-release phases of an emergency. The latter includes the transition between the short-term post-release and recovery phases (e.g. the first year(s)). , The project finishes at the end of 2019 with a final meeting in Bratislava, Slovak Republic. This paper presents the main findings from the different work packages.

The work-programme of CONFIDENCE was defined with the objective to understand, reduce and cope with the uncertainty of meteorological and radiological data and their further propagation in decision support systems. This included atmospheric dispersion, dose estimation, foodchain modelling and countermeasure simulations models. Consideration of social, ethical and communication aspects related to uncertainties was a key aspect of the project. Improvements in modelling and combining simulation with monitoring will ultimately help us to gain a more comprehensive picture of the radiological situation and will clearly improve decision making under uncertainties. Decision making principles and methods were investigated, ranging from formal decision aiding techniques to simulation-based approaches. These were demonstrated and tested in stakeholder workshops applying the simulation tools developed within CONFIDENCE. A comprehensive education and training programme was fully linked with the research activities.

Ensemble modelling for atmospheric dispersion using weather forecast ensembles and including source term uncertainties demonstrated the capability to describe uncertainties in the early phase of an emergency. Operational recommendations were developed for practical application of ensemble modelling in an emergency context. Improved monitoring approaches including for internal and external dosimetry were developed. Process based foodchain models were developed and compared with existing models that are based on transfer ratios. Stakeholder panels were set up to investigate the development of sensible countermeasure strategies and define important attributes for decision making from the local to national level. Historic events were analysed, exercises observed, mental models developed (theoretical frameworks to understand people's behaviour) aiming to better identify uncertainties that are important for decision-making. Multi-criteria decision tools were improved for uncertainty handling, thus being capable to digest now uncertain information from simulation models or uncertain preferences of decision makers. Key findings were integrated into a training framework that disseminated results to young scientists as well as professional in emergency preparedness and response.

The project also identified topics for further research as so far key uncertainties were identified, but the reduction of uncertainties is still at a very early stage.

Keywords: *Emergency Management, Uncertainties, Decision Support*

OS6.5 (T6.5-0056)

A software tool for radioactive waste management and remediation strategies comparison after a nuclear accident

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In nuclear post-accident management, a clear need has emerged to assess the impact of different remediation strategies on the amount and type of waste produced.

This observation has been pointed out as a major issue in the Fukushima Daiichi nuclear power plant accident, where the management of radioactive waste created by the implementation of multiple remedial actions (contaminated soil, water contaminated after cleaning…) has turned out very challenging.

The management of the radioactive waste aims to optimize their disposal, class by class, towards dedicated existing or future facilities. It depends directly on the waste amount, type and level of contamination to manage. The distribution of these amounts is determined by the remediation strategy selected, defined by a set of remedial actions to be implemented depending on location, contamination levels or land use.

To deal with this matter, a software tool named dewaX has been developed to assess the specificities of radioactive waste which could be generated due to the application of a given remediation strategy on a contaminated land. Thus, dewaX is able to compare different remediation strategies based on the following calculated outputs:

- the amount of waste produced after the application of a remediation strategy, class by class,
- the amount of agricultural waste (or agricultural losses), arising from the contamination of crop and livestock productions, produced after the application of a remediation strategy,
- the cost of the remediation strategy implementation,
- the workforce used during each remedial action,
- the deposition, radionuclide by radionuclide, within contaminated lands after the implementation of a remediation strategy.

All these outputs are geotagged, by using standard GIS files.

The purpose of these assessments is to provide rational information on the impact of remediation strategies and thus to support decision-making.

Keywords*: accident, waste, remediation*

OS6.5 (T6.3-0219)

A Path-planning Model Study for Emergency Evacuation under Nuclear Accidents

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Rescue and Evacuation is the important intervention strategy for the nuclear emergency under a serious nuclear accident such as Fukushima Accident. In order to avoid or reduce radioactive exposure, a reasonable path planning should be made and optimized in real time to adapt the change of radiation field caused by the changed wind field and releasing conditions. Virtual simulation is accepted as an effected research method due to the radiation hazard. The main purpose for study of the evacuation path planning is to find an optimal path in real nuclear response condition to accept the least dose, however, current studies are aimed at path planning without fixed existing roads. In this paper, the model of emergency evacuation path planning under nuclear power plant accident conditions is studied. Considering the situation of road resistance and dose rate distribution, an evacuation equilibrium model based on updated Dijkstra algorithm is proposed to plan the evacuation path, which reasonably arrange personnel evacuation, and ensure the minimum exposure of personnel during the evacuation process. In order to achieve accurate personnel exposure does, a Lagrangian nuclide diffusion model was applied in the calculation of dose rate distribution with the consideration of the wind direction change in real-time. In addition to that, the regulations intervention dose levels and intervention time was considered in the model to obtain combined path options the get the minimum accumulated collective does. The model was carried out at the platform of Virtual Nuclear Power Plant in Digital Society environment Virtual4DS developed by Institute of Nuclear Energy Safety Technology, Chinese Academy of Sciences •FDS team. In order to validate the effectiveness of the model, the path options gained from this work were applied to the scenario after the occurrence of the Daya Bay nuclear power plant accident. With the consideration of road condition, meteorological factor, and regulations does level, the combined path options set by the model make it possible to minimize the accumulated collective does during the whole process of the evacuation, and it can provide scientific support for emergency decision-making in the response of the nuclear accident, especially in the early stage.

***Keywords**: Nuclear Emergency, Evacuated Path Planning, Virtual Simulation, updated Dijkstra Algorithm*

OS6.5 (T2.2-0259)

Monte Carlo simulations of a CZT detector for post accidental dosimetry

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This work is focused on Monte Carlo (MC) simulations of a small CZT detector considering radiation or nuclear accidents. The CZT detectors seem to be convenient for on site/unmanned aerial vehicle measurements in these cases. Therefore, assuming possible emergency scenarios, the MC simulations were performed using the MCNP6.1 transport code.

A mathematical model of the detector was based on sketches from the manufacturer and X-ray images. The detector model was validated with a radium (^{226}Ra) needle and a standard ^{60}Co source. Differences between simulations and measurements did not exceed 4 %, therefore we accepted the model.

Afterwards, efficiency calibrations for selected industrial and medical sources were prepared, as well as calculations for semi-infinite surface contamination with artificial radionuclides. In situ measurement geometry at 1 m above the ground and measurements with drones at chosen heights (> 1 m) were supposed. For MDA values estimation, natural background spectra were simulated respecting specific activities of natural radionuclides in soil in the Czech Republic. Depending on the detector heights and considering acquisition time of 5 s, MDAs were in a range of several MBq to GBq for point sources, while in the case of surface contamination, MDAs were equal to hundreds of kBq m^{-2}.

***Keywords**: Monte Carlo simulations, CZT detectors, radiation/nuclear accidents*

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OS6.5 (T6.3-0646)

Detection of radioactivity of unknown origin: Inverse atmospheric transport modelling in the urgent phase of an emergency

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In the past decades, several radiological incidents were detected through the measurement of radioactivity in the atmosphere, whilst no (other) information was available about the incident. Such a detection of radioactivity of unknown origin falls within the Emergency Preparedness Category IV of the General Requirements described by the IAEA (IAEA, 2015) and requires to determine where and for whom protective actions and other response actions are warranted.

In that context, a method is presented that allows for the estimation of an unknown source term, i.e. the source location, moment of release, and the released quantity, purely based on measurements. The goal is to quickly provide information about the possible source and to assess whether and where (precautionary) urgent protective actions are necessary. These include evacuation, sheltering or taking an iodine thyroid blocking agent, to avoid or to minimize severe deterministic effects and to reduce the risk of stochastic effects. In practice, this means that the method should deliver results fast; within approximately one hour after collection of the measurements.

The method is based on the adjoint source-receptor relationship (Pudykiewicz, 1998). The adjoint transport equation is solved by using a modified version of the atmospheric dispersion model NPK-Puff (Eleveld, Kok, & Twenhöfel, 2007), (Tomas, van Dijk, & Twenhöfel, 2017) in which time and advection are reversed. In addition, the monitoring stations that detect elevated levels of the ambient dose equivalent rate are used as the sources of the adjoint concentration (Peereboom, 2019). The method is applied to two synthetic cases, a continuous straight wind situation and rotating wind situation, for which the effect of the density of the monitoring network on the predicted source term is assessed. Subsequently, the method is used to predict the source term of the European Tracer Experiment 1 (ETEX1) (Van dop et al., 1998). ETEX1 was an experiment carried out in 1994 in which a tracer gas was released in the west of France, which was measured by a network of 168 ground-level sampling stations spread over a large part of Europe. By considering this actual release the effect of discrepancies between the numerical weather forecast data and the actual meteorology are included. This test case is therefore representative of a radioactive release with an affected area at a continental scale.

For each possible source location, the maximum correlation in time is determined between the adjoint concentration released from each monitoring station and the tracer measurements at the monitoring stations. The dispersion calculation comprises a single simulation (with multiple sources), which can be completed within one hour. For the ETEX1 case the estimated location is within 60 km from the actual location and the estimated released time corresponds to the end of the actual release interval. Finally, the estimated released quantity is within a factor of two of the actual release.

Concluding, the results show that a preliminary estimation of the source term, and subsequently the assessment whether urgent protective actions are required, is possible in the urgent phase of an incident.

Keywords: Inverse modelling, source term, emergency response

OS6.5 (T6.3-0646)

Detection of radioactivity of unknown origin: Inverse atmospheric transport modelling in the urgent phase of an emergency

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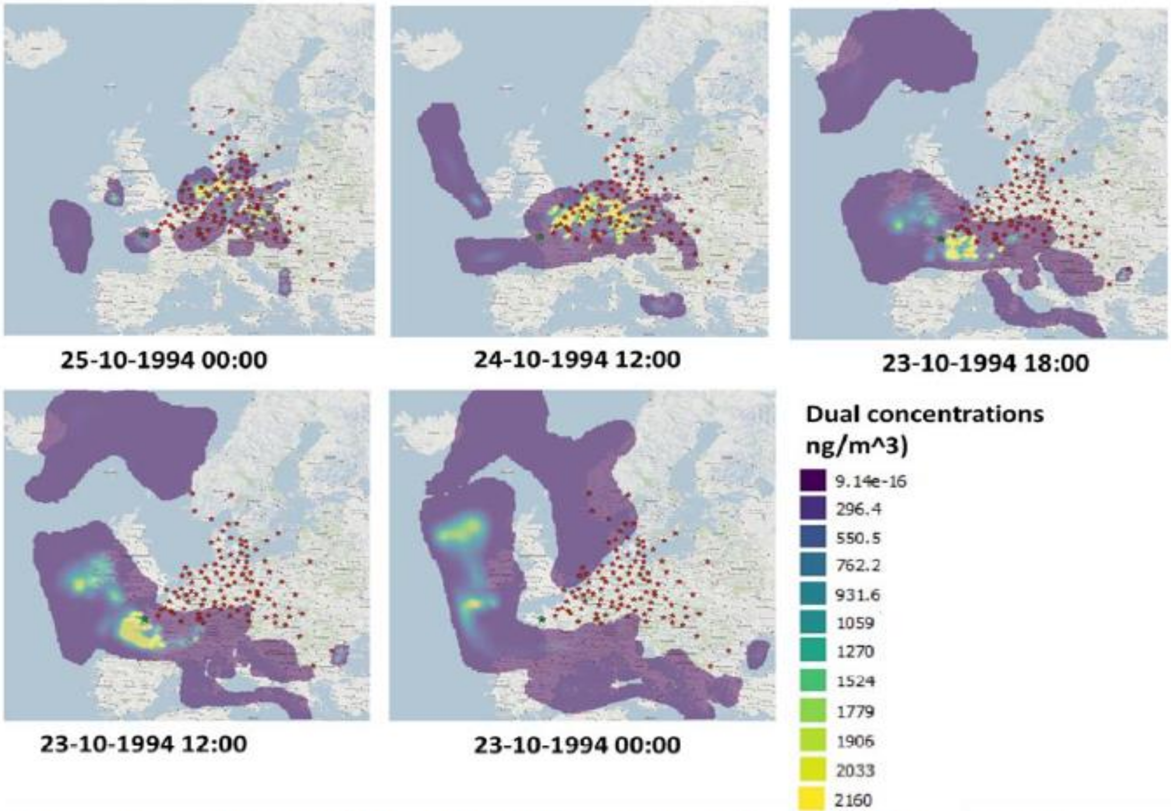

Figure 1: Adjoint ("dual") concentration released from monitoring stations (red stars) at five moments in time for ETEX1. The green star (west of France) is the source location of the tracer. From left to right: 25/10/1994 0:00, 24/10/1994 12:00, 23/10/1994 18:00, 23/10/1994 12:00, 23/10/1994 0:00 UTC.

OS7.1 (T7.5-0241)

Environmental assessment of the areas contaminated as a result of the nuclear accident at Chazhma Bay

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A serious nuclear accident occurred on August 10, 1985, during scheduled operations with the nuclear submarine in the dockyard (Chazhma Bay, Primorsky Krai). It resulted in the release of artificial radionuclides in the environment and contamination of marine and terrestrial environment. The main long-lived radionuclides were ^{60} Co (65–90%), ^{54}Mn (4–16%), and ^{137}Cs and ^{90}Sr (less than 1%). Radioactive fallout caused contamination of adjacent forest area with the formation of "cobalt radioactive trace". The modern status of the contaminated area is classified as a nuclear legacy site and the public exposure refers to an existing exposure situation. At present, the local population uses these territories occasionally to pick up berries and mushrooms. The aim of this study is to characterize the contaminated areas after thirty years of the accident. The number of surveys was conducted at the "cobalt radioactive trace" during 2015-2019. ^{60}Co, ^{137}Cs and ^{235}U are the main artificial radionuclides presented in the soil of contaminated forest areas. The local ^{60}Co contaminated areas were identified where the dose rate exceeds the background values several times (from 0.1 to 0.6 μSv/h), and the ^{60}Co content in the soil reaches 5000 Bq/kg. The study of the environmental behaviour of the radionuclide reveals its predominant concentration in the top layer of the soil (0-5 cm) and low mobility in the environment. Cs^{137} and U^{235} activity concentrations exceed regional background values up to 10 and 20 times respectively and do not have a significant impact on the radiation situation. It is shown that the study area is characterized by local areas of soil contamination with heavy metals (V, Pb, Mn) exceeding the established permissible levels. The obtained results were used to assess the public doses within the localized contaminated areas and to determine the need for protective measures.

***Keywords**: submarines, nuclear legacy sites, cobalt trace,*

OS7.1 (T7.5-0365)

From 300,000 Bq/l to 300 Bq/l: Remediation of tritiated groundwater at the High Flux Reactor in Petten, the Netherlands

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After finding enhanced activity concentrations of tritium in groundwater at the High Flux Reactor in Petten, the Netherlands in September 2012 an investigation was started to find the cause, the extent of the contamination and to prevent further dispersion.

The removal of tritiated groundwater was done in two phases: first, the removal of so-called 'hot spots' (activity concentrations above 10 kBq/l), starting in February 2013, and from June 2014 up to now the final remediation phase. For these activities a license was issued by the competent authorities, based on an existing exposure situation.

Groundwater dispersion was modelled and the effect of pumping up groundwater was taken into account to adjust the model. The effect of removing groundwater seemed less effective than the first model predictions on which the intervention license was issued. Therefore, in 2018 a new plan of approach was made and a license change application was issued in which it was requested to raise the intervention limits (maximum allowable activity concentration for tritium). On one hand it was hard to find a justification to invest in remediation in order to prevent a trivial dose for members of the public, and on the other hand it was hard to convince the public of the insignificant risks of the residual amount of tritium in groud- and surface water.

According to the new model predictions, in Spring 2019 tritium was found in surface water adjacent to a tulip bulb field. In follow-up measurements tritium concentration were below the MDA of 10 Bq/l. Due to timely and open communication public response was 'mild'.

This paper will discuss the lessons learned so far regarding the process of remediation, the effectivity, and the impact on normal operations, but also the licensing aspects and last but not least the communication strategy with members of the public. Between 2012 and 2019 tritium activity concentrations were reduced from 300,000 Bq/l to about 300 Bq/l.

Keywords: *tritium contamination, existing exposure situation, groundwater, remediation, intervention, communication strategy*

OS7.1 (T7.2-0368)

Radiometric Investigation of Beach sand in Zanzibar, Tanzania

Since time immemorial, human beings have always been exposed to radiation from cosmic rays and from the earth's crust. Their levels and distribution pattern depends upon the local geology of each region of the world, geographical conditions, transport processes as well as sediment formations. Despite the interest of the scientific world to assess the exposure to terrestrial radiation and radionuclides, there is no data regarding terrestrial radiation about Zanzibar until now. The coastline of Zanzibar is among the tourists' destinations in East Africa. The coastline is also used for human settlement and other activities such as fishing and subsistence farming.

The radiometric study has been conducted in Zanzibar beaches using a combination of ex-situ and in-situ gamma-ray spectroscopy. The in-situ gamma-ray survey was conducted using the **M**ulti **E**lement Sediment **D**etector for **U**nderwater **S**ediment **A**ctivity (MEDUSA) detector. The detector was mounted on the front of a 4×4 vehicle, 60 cm off the ground. Activity concentrations of the primordial radionuclides were extracted from the MEDUSA spectra using the **F**ull **S**pectrum **A**nalysis (FSA) procedure. The collected beach sands were analysed ex-situ using the low background **H**yper-**P**urity **Ge**rmanium (HPGe) detector system.

The activity concentration levels for ^{232}Th, ^{238}U and ^{40}K and radiological hazards indices have been evaluated. From the study, it was found that in most beaches, the activity concentration levels are much lower with one major exception at Kukuu beach, found to have enhanced radioactivity levels due to the presence of heavy minerals. The highest absorbed dose rate found in this beach is 38 times higher than the average world level of 57 nGyh^{-1} for terrestrial doses. Apart from the samples from Kukuu beach that contain the high ^{238}U and ^{232}Th levels, the beach sands in Zanzibar do not pose any radiological threat to visitors and the public using beaches for various activities.

OS7.1 (T7.4-0101)

Preliminary Assessment of Naturally Occurring Radionuclides and Associated radiological impact of Tantalum Mining at Bikita Minerals in Zimbabwe

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A radiological survey was conducted at Bikita Minerals in Zimbabwe to ascertain baseline radioactivity levels of naturally occurring radioactive materials (NORM) and external gamma exposure risk to workers. Direct gamma spectrometry was used to determine the concentrations of Uranium-Thorium series radionuclides and ^{40}K in the processing and separation of the Tantalum ore. The study also evaluated the absorbed dose, annual effective dose from external gamma dose rate, radiological hazard and associated cancer risk. The average activity concentrations of ^{238}U, ^{232}Th and ^{40}K in the ore were 54.17, 24.37 and 627.20 Bqkg^{-1} respectively. The average activity concentrations of ^{238}U, ^{232}Th and ^{40}K in the tailings were 124.59, 13.67 and 505.24 Bqkg^{-1} respectively. The values were generally above the worldwide average activity concentrations with exception of ^{232}Th. Annual effective dose due to external gamma exposure at the Production Plant- wet concentrated ore section and the Storage warehouse was 1.39 and 7.94mSv respectively, which is above the IAEA and ICRP recommended level of 1mSv/year for existing exposure situations. The average absorbed dose rate in air due to the radionuclides in the ore and slimes was 78.85nGy/h. This was above the world average value of 60 nGy/h. Consequently, the calculated average outdoor annual effective dose for the study area was 0.097mSv/y, which is slightly above the world average value of 0.07mSv/y but falls within the recommended limit of 1mSv/y. The study showed that Tantalum mining at Bikita Minerals Mine significantly elevates radiation levels thus posing a radiological hazard to workers.

OS7.2 (T7.1-0496)

Planned approach for indoor radon mapping in the West Rand Region, South Africa

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The West Rand area is dominated by abandoned tailings dams from gold and uranium mines deposited from a century old mining, which could be the potential source for an elevated radon level in the environment. Radon gas has long been identified as a human carcinogen and accounted to be the second largest cause of lung cancer after smoking. The purpose of this study was to understand the occurrence and distribution mechanism of radon gas in the West Rand region. Also, to investigate whether the radon gas released from uranium-bearing geological formations and mining sources result in significant health effects such as lung cancer in the area. In this study, the sampling of rocks, tailings and construction materials was conducted for geochemical analyses. For characterisation of radon, 60 radon monitors were installed in indoor and outdoor environments.

The results showed that mine tailings contained high uranium levels, with a maximum of 149.76 ppm and a mean value of 48.87 ppm. The radon levels in the area ranged from 31.7 Bq/m^3 to 1068.8 Bq/m^3 and thus immensely exceed the typical expected outdoor radon level of about 10 Bq/m^3. Significantly high average values of 187.4 Bq/m^3 were obtained from gold tailings dams. The radon distribution was mainly controlled by wind and distance from the tailings. In indoor environments, radon concentration ranged up to a maximum of 173.5 Bq/m^3, which is above the 100 Bq/m^3 recommended by the World Health Organization. The effective doses received by the public showed a maximum of 15.50 mSv/y, which is above the recommended value of 1 mSv/y proposed for public exposure. The estimated doses have a greater potential to pose a high health risk to the exposed populations. Corroborating the aforementioned statement, a high frequency of deaths that are related to lung cancer is recorded in the area associated with the elevated radon levels.

Keywords: *Abandoned mine tailings dams, Lung cancer, Radon.*

Oral Session T7

OS7.2 (T7.1-0130)

Radon Protection in Appartments by Wireless Radonactivity Concentration-Controlled Ventilation

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The new German Radiation Protection Act (StrlSchG) of 31 December 2018 established a reference value of 300 Bq/m^3 for the annual average radon activity concentration in buildings with recreation and living rooms as well as workplaces. It is expected that the reference value will be exceeded in a vast number of buildings throughout Germany and that radon protection measures will become indispensable. A simple, efficient and cost-effective radon protection measure for existing buildings is ventilation. In the scope of a joint project, ventilation systems with zone control are to be extended by the control parameter radon activity concentration. A radon monitor will be developed for this purpose, which can be integrated wireless into ventilation systems. Three-week radon measurements were carried out in 13 apartments of an uninhabited apartment block. High radon activity concentrations were found in all three floors. The radon activity concentrations depended on the outside temperature and heating regime. The maximum values were 14,700 Bq/m^3 on the ground floor, 6,000 Bq/m^3 on the first floor, and 2,000 Bq/m^3 on the second floor, respectively. Ventilation experiments were carried out in an apartment with high radon activity concentration. Two decentralized ventilation systems with heat recovery were installed in each of the two opposite outside walls. The controlling device of the system was activated wirelessly depending on the radon activity concentration. The radon activity concentration was reduced from 8,000 Bq/m^3 to 800 Bq/m^3 in a first experiment in the living room (Dehnert et al. 2019). Ongoing research is directed to the optimization of the system to ensure permanent undercut of the reference value.

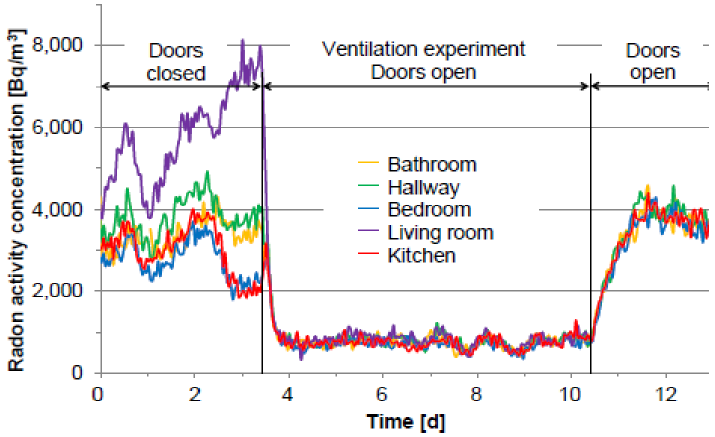

Fig. 1. Course of the radon activity concentration during a ventilation experiment in an appartment

Keywords: *radon, radon in homes, decentralised ventilation*

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OS7.2 (T7.1-0163)

Evaluation of the effect of ion generation and modulation of atmospheric characteristics on the hazard from radon and its progeny

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The hazard associated with radon is primarily from its progeny. Traditional remediation focuses on the removal or preclusion of radon gas from a volume of air and can be unfavourable due to cost and practical considerations. In previous studies ion generation has been shown to reduce the equilibrium factor but increase the unattached fraction. This work quantifies the total effect of ion generation on overall dose from radon progeny in an indoor scenario, and also evaluates the efficacy of combining ion generation with modulation of other atmospheric characteristics.

Keywords: *radon, progeny, remediation*

OS7.2 (T7.1-0202)

Seasonal variations of radon concentration in workplaces of GA-East District of Greater Accra Region of Ghana

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Keywords: *offices, laboratories, CR-39*

Indoor radon concentration in buildings with respect to two main seasons as rainy and dry weather conditions, have not yet been evaluated in Ghana. At the present time, Ghana does not have national guidelines specifying the acceptable radon levels in workplaces but have been using international action levels and limits. Most studies on radon in Ghana have focused few studies on water, soil and buildings but no workplaces. The Ghana Atomic Energy Commission (GAEC) houses numerous laboratories and offices with staff strength of almost 700 people. Due to the eight (8) hours or mores daily working schedule, for most of the staffs in Ghana, one can classify the offices as "long occupancy" sites. This necessitates the importance of carrying out indoor radon measurements at these offices and laboratories together with some residential communities within the region for an adequate risk assessment. This study uses one hundred and forty four (144) CR-39 detectors were used for indoor radon measurements of which 118 offices and 26 laboratories for January-December and seasonal studies from June-August, December-February and March-May respectively. Indoor radon concentrations for the studied buildings were randomly selected within the premises of GAEC and other offices in Ga-East District of greater Accra region of Ghana. In each building, a detector was placed either in an office or laboratory, at a height of 1 to 1.5 m above the floor, at a distance greater than 0.5 m from each wall, and at a minimum of 15 cm from any other objects for period of 12 months and 3 months. Exposed, radon detectors were sent to Centre for Radiation Protection, Regional Agency for Environmental Protection, Friuli Venzia Giulia, Udine, Italy for the analysis. The detectors were etched and latent tracks formed were counted in 144 fields using an optical microscope of 40 × magnification objective lens. The tracks density left on track films were then used to evaluate the indoor radon concentration. Normality associated with the result was determined using cumulative frequency distribution and other statistical analysis. June-August was found to contain high radon concentrations than the other periods in all the studied locations. Average radon exposure in the offices and the laboratories were found to be 129.8 Bq/m^3 and 173.7 Bq/m^3 higher than the concentrations. In general, laboratories had radon concentration greater than offices. The studied result ranged 45.6 - 558.7 Bq/m^3 of which 28.3 were found to be greater than action levels proposed by WHO, indicating that not all the buildings in the studied area is negligible. The results were compared with similar studies done in Ghana and other Radiation Safety Standards and international limit. The result from the studies will be very important in formulating guidelines for radon exposure map and strategy for the control of radon exposure in workplaces in GA-East District of greater Accra region.

Acknowledgement:

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OS7.2 (T7.1-0317)

Inhalation Dose Assessment From ^{222}Rn & ^{220}Rn In Extremely High Background Radiation Area, Indonesia

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Several countries in the world have High Background Radiation Area (HBRA) with potential public annual effective non-fractionated doses from external and/or internal exposures, which are even higher than 20 mSv y^{-1} dose limit for radiation workers. Some studies in the HBRA areas of Brazil, China, India, and Iran have received more attention in recent years. Recently, there is also an HBRA area in Indonesia, namely Mamuju, and this area has high exposure to natural radiation both internal and external exposure. Radon gas contributes to the highest level of natural radiation dose and generally comes from the soil, water, or the building materials with natural radioactive elements namely Radium (^{226}Ra), the decay product of Uranium (^{238}U). These radioactive elements decay by emitting an α particle to generate ^{222}Rn and can cause cancer. Therefore, investigation about inhalation dose assessment from ^{226}Ra in HBRA at Mamuju is needed as a radiation protection to general public.

This study activity has several measurements including measurements of radon (^{222}Rn) and thoron (^{220}Rn) using Raduet® monitor (Radosys. Ltd, Hungary) as Solid State Nuclear Track Detectors (SSNTDs) named CR-39 with two-month exposures period in one year and also radon-thoron progeny monitor to be installed in 450 houses, which includes 100 houses in control areas, 100 houses in medium radiation areas and 250 houses in high-radiation areas, where high radiation areas cover 5 districts (Botteng, North Botteng, Takandeang, Ahu and Taan). Besides that, measurements using active monitor (RAD7, Durridge Ltd, USA) for three-day continuous measurement to determine the daily concentration of ^{222}Rn & ^{220}Rn both indoors and outdoors. ^{222}Rn concentrations in dwellings have ranged from 130 – 1,929 Bq m^{-3} and ^{220}Rn concentration ranged from 105 – 2,253 Bq m^{-3}. Then, the numerical model-IMBA Professional Plus (Integrated Modules for Bioassay Analysis) simulation (developed by Public Health England, UK) approach for internal dosimetric was conducted to get an inhalation dose assessment.

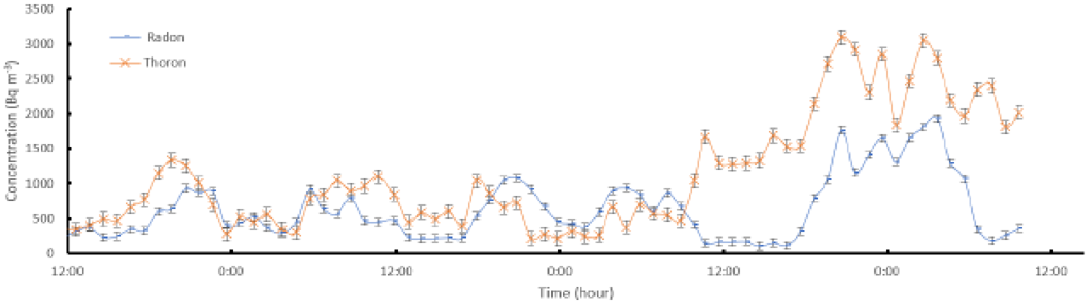

Fig. 1. The daily concentration of radon and thoron

Keywords: HBRA, inhalation, radon&thoron

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OS7.2 (T7.1-0325)

Calibration chamber for radon, thoron and their progenies at Hirosaki University, Japan

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Recently, the ICRP has recommended new dose conversion factors for radon (Rn) which corresponds twice as the previous value[1)]. The ICRP has also mentioned dose conversion factors for thoron (Tn). Therefore, it is more important to evaluate an inhalation dose from Rn, Tn and their progenies than before. The devices used for measurements of Rn, Th and their progenies must be calibrated for QA/QC. However, there is a few calibration facilities especially for progenies in the world. The calibration chamber for Rn and Tn were established at Hirosaki University[2)]. In the present study, the calibration chamber for Rn and Tn progenies were designed by applying the Rn and Tn calibration chambers, and the performance tests were carried out.
The Rn and Tn calibration chambers are composed of four parts; a source unit, a mixing chamber, an exposure chamber and a monitoring unit. Natural rock samples and lantern mantles are used for Rn and Tn sources, respectively. Rn gas generated from the source is transferred to the exposure chamber through the mixing chamber by an air pump. In case of Tn gas, the mixing chamber was used for the exposure chamber because of its short half-life. It was found that Rn and Tn concentrations could be controlled in the range of 200-1,000 Bq m^{-3} and 3,500-30,000 Bq m^{-3}, respectively. In order to establish a calibration chamber for Rn and Tn progenies, some aerosol generators were used with NaCl solution, as shown in Fig. 1. The aerosol generator is selected according to the aerosol size. Aerosols and the progenies were mixed in the mixing chamber to make radioactive aerosols, then the radioactive aerosols were injected to the exposure chamber. The aerosol number concentration and the aerosol size distribution are measured by a Scanning Mobility Particle Sizer and a Laser Aerosol Spectrometer, respectively.
As a result of the performance test, it was found that the aerosol number concentration could be controlled by changing the flow rate of the aerosol generator in the range of 800-230,000 cm^{-3}. The aerosol size distribution also could be controlled by the concentration of solution in the range from 60 to 113 nm of median diameter. Equilibrium Equivalent Thoron Concentrations measured by a grab sampling with Si semiconductor detector were stabilized at least 12 hours. The detail results will be presented in this presentation.

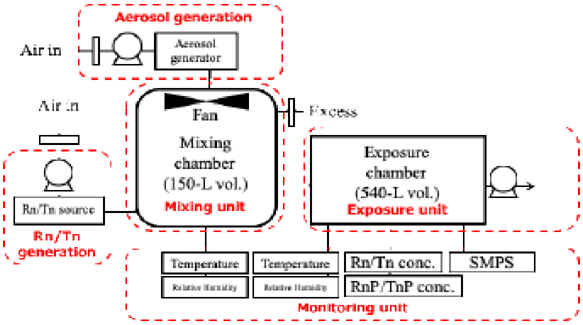

Fig. 1. Schematic diagram of the calibration chamber for Rn and Tn progenies.

Keywords: *Radon, Thoron, Radioactive aerosols, Calibration*

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OS7.3 (T7.4-0321)

Radioactive Investigation Management Database Designing for NORM

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From the available results of some radiological surveys for NORM industry in international organizations, other countries and countries, the range of industries involved about NORM is very wide. In order to improve efficiency of analysis, reduce human error and other considerations the NORM radioactive waste in the provinces, industries, waste types, waste volume and related processes and more, in this paper briefly introduces the national survey data system designing analysis. The system could solve the above-mentioned problems and reduce the statistical workload of the procedure. Users only need to input, the relevant basic data once, then the system support to analysis information, data and support information reuse capabilities. The data system could save time and improve the reliability of investigations.

Keywords: Radioactive; NORM; Natural Radiation; Radiation; Pollution; Emissions;

OS7.3 (T7.4-0129)

Gamma spectroscopy assessment of naturally occurring radioactive materials in and around Mupani Goldmine along the Tati-Green Belt in Botswana

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Study has been carried out at Mupani Gold-mine and its surroundings along the Tati Green Belt in Botswana to determine the exposure to both the mine workers and the public from naturally occurring radioactive materials due to mining and processing of gold ore. Direct gamma spectrometry was used to analyze 43 soil samples from the mining environment and 21 soil samples from outside the mine as control using HPGe detector. The results have shown that, from the mine, the U-238 activity concentration ranges between 6.306 Bq/kg to 47.174 Bq/kg, with the average value of 21.803 Bq/kg, the Th-232 activity concentration ranged between 2.789 Bq/kg to 28.344 Bq/kg, with the average value of 12.717 Bq/kg, while the K-40 activity concentration ranged between 21.970 Bq/kg to 455.900 Bq/kg, with the average value of 204.001 Bq/kg. For control samples, the results have shown that, the U-238 activity concentration ranged between 7.490 Bq/kg to 31.572 Bq/kg, with the average value 17.722 Bq/kg, the Th-232 activity concentration ranged between 7.278 Bq/kg to 22.609 Bq/kg, with the average value of 13.887 Bq/kg, while the K-40 activity concentration ranged between 89.860 Bq/kg to 472.200 Bq/kg with the average value of 222.62 Bq/kg. Radiological hazard parameters calculated from these activity concentrations were lower than the recommended safe limits. Calculated averages values for the external hazard (H_{ext}) and the internal hazard (H_{in}) from the mine were found to be 0.15 and 0.21 respectively. Both these values are lower than unity, thereby, posing no health risk to the mine workers and the population in the mining area.

Keyword: *Tati-Green Belt; gamma spectroscopy; activity concentration; radiological hazard*

ACKNOWLEDGMENTS

The author (first author) would like thank the University of Botswana for financial support during the research period. Also thankful to the following institutions for technical assistance: Mupani Gold Mine for sample collection; Botswana Geological Institute for soil sample preparation to the required standard and specifications; The Centre for applied radiation science and technology with gamma spectrometry analysis.

OS7.3 (T7.4-0492)

Radioactivity and toxicity in ground and surface water around the Shiva uranium mine in Klerksdorp, South Africa

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Uranium mining is an important aspect of the nuclear fuel cycle, providing the fuel used in nuclear power plants. Uranium-238 occurs in nature together with its daughters. During the mining of uranium, the daughter nuclides are not extracted and therefore are dumped in mine tailings, where they may present a risk to both the environment and the public living near the mining operations. This study was carried out at and around the Shiva uranium mine, with the aim of determining the effect of the mining activity on the water sources around the mine. Water samples were collected and analysed to determine the activity concentration of natural radionuclides. Samples were analysed using Liquid Scintillation Counting for gross alpha/beta activity, ICP-MS for uranium, thorium and other heavy metals, gamma spectrometry using an HPGe detector.

The activity concentration in water samples collected from Bobgunner dam and Dominionville were analysed using ICP-MS to determine uranium isotopic ratios and total uranium, the isotopic ratios were used to determine the individual isotope concentration from the total concentration. Thorium and potassium were also determined. The average activity concentrations were 2.83 Bq/L, 0.14 Bq/L, 3.26 Bq/L, $\geq$0.0003 Bq/L and 0.03 Bq/L for Bobgunner dam and 413.58 Bq/L, 19.23 Bq/L, 409.58 Bq/L, 21.15 Bq/L, $\geq$0.0003 Bq/L for Dominionville for ^{238}U, ^{235}U, ^{234}U, ^{232}Th and ^{40}K respectively. The activity concentrations were compared to the world average concentrations determined by World Health Organization, 2008 of 10 Bq/L, 1 Bq/L, 10 Bq/L for ^{238}U, ^{234}U and ^{232}Th respectively. The activity concentrations were also compared with the activity concentrations determined by DWA limit values of 0.89 Bq/L, 0.228 Bq/L and 50 mg/L for ^{238}U, ^{232}Th and ^{40}K respectively. The activity concentration of ^{238}U, ^{234}U, ^{232}Th and ^{40}K was found to be higher than the WHO values for the water samples from Dominionville except for thorium and for Bobgunner dam they were slightly below the WHO limits. The concentration of the toxic elements was 1,04E-03 mg/L, 4,71E-04 mg/L, 2,27E-05 mg/L, 1,08E-03 mg/L, and 2,29E-01 mg/L for Bobgunner dam and 6,00E-04 mg/L, 2,22E-03 mg/L, 9,12E-01 mg/L, 5,24E-01 mg/L and3,32E+01 mg/L for Dominionville for Hg, Pb, Cd, As, Co, and U. respectively. The concentrations were also compared with the concentrations determined by DWA guideline of 1000 mg/L, 10000 mg/L 0.005 mg/L, 10000 mg/L and 1 mg/L for Hg, Pb, Cd, As and U respectively. The results showed that activity concentration values of all the water samples from Bobguuner dam are within recommended limit values and for Dominionville samples are above are above the limit values by the WHO guideline. The toxic elements for samples from Bobgunner dam were within the recommended limit values and for Dominionville they were above the limits for toxic elements by Department of water affairs (DWA).

Keywords: Water contamination, Radiological risk, NORMs

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OS7.3 (T7.4-0548)

ASSESSMENT OF HUMAN HEALTH EXPOSURE FROM CONTAMINANTS ASSOCIATED WITH WASTE GENERATED FROM GOLD MINING

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Mining activities has been identified as a source of heavy metal contamination of the environment as well as exposure to naturally occurring radioactive materials (NORM). In Ghana, there are over two hundred (200) registered mining companies operating small, medium and large scale mining. It is also estimated that over five thousand (5000) people are engaged in illegal small-scale mining across the country who carry out these activities with no supervision from appropriate National regulatory bodies. The activities of these small scale mining industry are getting more destructive as second-largest source of pollution after agriculture in Africa. Several serious health problems can develop as a result of excessive uptake of dietary heavy metals and natural radionuclides of which some are soluble in water and have the tendency to leach into water bodies and farmlands. The general aim of the studies is to assess the human health risk due to intake of heavy metal and natural radionuclides via consumption of food and fish as a result of the mining activities.

It is worth noting that, knowledge of pollution levels and trends is essential to define environmental protection measures and science-based policy for a sustainable waste management of the ecosystems and resources. Ghana is also in the process of formulating guidelines on setting standards for the regulation of NORM in the mining industry.

The availability of data from such studies is very vital to all stakeholders involved since it will add to the data required for the development of guidelines for the regulation of NORM in Ghana.

Accurate data on chemical constituents as well as other radioisotopic signatures of waste generated from mining activities is of paramount importance to carry out any potential human health risk assessment. These data will be useful not only to evaluate the present radiochemical state of the environment, but also for subsequent evaluations of the possible future environmental contamination and bioaccumulation due to activities of the extractive industry.

Keywords: mining, waste, human health risk assessment

OS7.3 (T7.2-0594)

Population exposure to natural radiation in Betare-Oya, gold mining areas, eastern Cameroon

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The objective of the study is to investigate the natural radiation exposure to the public in gold mining areas of Betare-Oya and vicinity. For this purpose, the car-borne survey using a 3-in × 3-in NaI(Tl) scintillation spectrometer was used to determine activity concentrations of ^{238}U, ^{232}Th and ^{40}K in soil and air absorbed dose rate contour map for a detailed evaluation of external dose. For the internal dose, indoor radon (Rn), thoron (Tn) and thoron progeny (TnP) were simultaneously measured using RADUET detectors and TnP monitors. The average values of activity concentrations in soil of ^{40}K, ^{238}U and ^{232}Th were 197 ± 21 Bq kg^{-1}, 37 ± 13 Bq kg^{-1} and 32 ± 7 Bq kg^{-1} respectively. Absorbed dose rates in air ranged from 23 to 80 nGy h^{-1} with a mean value of 44 ± 7 nGy h^{-1}. Rn and Tn concentrations ranged between 88–282 and 4–383 Bq m^{-3}, respectively, with the arithmetic means of 133 ± 39 and 93 ± 76 Bq m^{-3}. The 76% of houses for Rn and 25% for Tn exceed the WHO reference level of 100 Bq m^{-3} and 3% of the houses exceed the ICRP threshold of 300 Bq m^{-3}. External annual effective dose ranged from 0.17 to 0.60 mSv y–1 with a mean value of 0.33 ± 0.05 mSv y^{-1} and the total inhalation dose due to Rn, Tn and their progeny ranges between 1.8 and 6.2 mSv y^{-1} with the arithmetic mean of 3.8 ± 1.1 mSv y^{-1}.

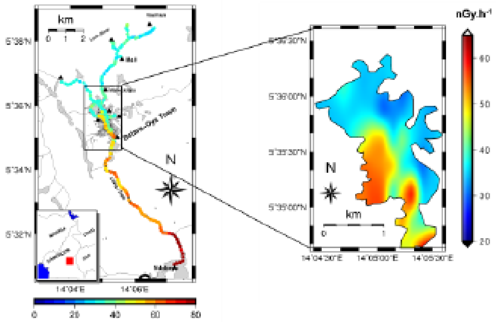

Fig. 1. Absorbed Dose rate in Air distribution (nGy h^{-1}) for all study area and detailed Absorbed Dose rate in Air distribution (nGy h^{-1}) of Betare-Oya town

Keywords*: Air absorbed dose rate, external dose, inhalation dose*

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OS7.3 (T7.4-0644)

Protocols and computational methods to support stakeholders of NORM industrial sectors in Italy: first results of a research project

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Taking into account the list of NORM industrial sectors considered in the EU BSS, the project is designed to develop technical and scientific tools useful to draft guidelines and procedures addressed to stakeholders at different levels involved in the implementation of the EUBSS as regards the Italian NORM industrial sectors. In the frame of the project, protocols of sampling and measurements of NOR materials are elaborated; moreover, procedures and computational methods are developed for estimating the dose of exposed workers and members of public.

In a first phase, particular categories of NORM industries of potential radiological impact for workers and members of the public, or having a strategic and/or economic importance for Italy, will be selected to study the relevant production processes, to characterize the NOR materials more representative of production phases. The characterization will be performed using the most consolidated radiometric techniques, in field and in laboratory. It is also planned a combined use of *in situ* characterization techniques, such as HpGe gamma spectrometry, and Montecarlo models: indeed, the *in situ* techniques and the Montecarlo models are already widely used for the characterization of waste, but within the project they will be adapted, where possible, to the selected plants. The development of this methodology is aimed at identifying the parts of the plants that may present significant levels of radioactive contamination. Based on the experience and data collected, for the different selected industrial sectors protocols to sample and characterize NOR materials will be drafted.

In the second phase, once the most significant exposure scenarios have been identified, calculation methods will be developed for dose estimation for workers and members of public. These tools will require the use of existing and well tested calculation codes, and the development of a dedicated user friendly software.

In the third phase of the project, attention will be focused on ensuring that the project deliverables the widest usability and dissemination among stakeholders, such as undertakers, citizens, public institutions and professionals (e.g. local competent authorities and radiation protection experts).

Keywords: *NORM, radiological characterization, dose assessment.*

ACKNOWLEDGMENTS

This project is supported by the National Institute for Insurance against Accidents at Work (INAIL) .

OS7.4 (T7.1-0042)

From radon and thoron measurements, inhalation dose assessment to national regulation and radon action plan in Cameroon

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The present work reports measurements of activity concentrations of ^{222}Rn, ^{220}Rn and its progeny in dwellings followed by inhalation dose assessment to the public in mining and ore bearing regions of Cameroon. These measurements were carried out from 2014 to 2017 using RADUET detectors and thoron progeny monitors for radon, thoron, and thoron progeny in 450 dwellings. As summarized in Table 1, it was found that on average 43% and 37% of houses in the study areas have radon and thoron concentrations above 100 Bq m^{-3}, respectively while 1% and 3.3% of houses have radon and thoron concentrations above 300 Bq m^{-3}. The contribution of thoron to the inhalation dose due to radon and thoron exposure ranges between 12-67, 3-80, 7-70, and 7-60 % in Poli, Lolodorf, Betare-Oya, and Douala. The corresponding average values are 49, 53, 31, and 26 %. Thus thoron cannot be neglected in dose assessment to avoid biased results in radio-epidemiological studies. It would be advisable to define action levels for thoron at the international level for radiological protection of members of the public. At the nationwide scale, after the above indoor radon survey the regulation on radon exposure and the radon action plan were drafted under the project on "Establishing a national radon plan for controlling public exposure due to radon indoors" developed within the framework of the Technical Cooperation between the International Atomic Energy Agency (IAEA) and Cameroon for 2018-2019. They will be adopted before mid-2020.

Table 1: Percentage of dwellings having radon (Rn) and thoron (Tn) concentrations above the reference levels of 100 and 300 Bq m^{-3} recommended by WHO and ICRP.

Study area	Rn>100 Bq m^{-3}	Tn>100 Bq m^{-3}	Rn>300 Bq m^{-3}	Tn>300 Bq m^{-3}
Poli	20	42	0	0
Lolodorf	47	54	1	10
Betare Oya	76	25	3	3
Douala city	27	27	0	0

Keywords: *Radon/thoron, inhalation dose, radon action plan*

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OS7.4 (T7.1-0570)

Update of WHO's survey of national radon-related policies and regulations

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In 2006, WHO conducted a survey among its Member States on their radon-related activities [1]. The collected data contributed to the development of the landmark WHO *Handbook on Indoor Radon* [2], published in 2009.

Based on strong scientific evidence on the increased risk of lung cancer from exposure to radon, WHO together with seven other international organizations proposed in 2014 specific requirements for public and occupational exposure to radon in the ionizing radiation *International Basic Safety Standards* [3]. A number of countries are reviewing their radon policies and regulations. In particular, most countries from the European Union were to develop radon-related regulations by February 2018, as per the mandatory transposition of the *EURATOM Directive* [4].

Given these regulatory revisions, WHO is updating its radon database via a new survey gathering information from its Member States on the status of radon policies and regulations in both residential and work settings. The online survey covers six specific topics:

1. National radon activities
2. National radon regulations and action plans
3. National radon reference levels
4. Radon concentration measurements
5. Radon exposure prevention and mitigation
6. Radon-related communication and linkages to other public health national strategies

An analysis of the survey results will be presented at the IRPA15 Congress. The data will later be published in the WHO Global Health Observatory, a website established to share data on global health, including member state statistics and information about specific diseases and health measures.

***Keywords**: Radon, Regulations, Survey*

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OS7.4 (T7.1-0181)

Overivew of Indoor Radon Research conducted in South Africa: 1980s to date

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The history of research on radon and indoor radon measurements in South Africa (SA) dates back to the mid 1980s and early 1990s. Small-scale studies have been performed in areas where high radon concentrations were expected due to the geology and the mining history of the area. Most of these studies were performed in Gauteng and Western Cape provinces. Gauteng province has a long history of gold mining and this has left behind large amounts of waste that contain long-lived naturally occurring radionuclides such as uranium-238 (^{238}U) which decays into radium-226 (^{226}Ra) from which radon-222 (^{222}Rn) emanates. Radon research conducted in the Western Cape province was mainly due to the geology of this region, which is rich in granitic rocks. This study reviews published journal articles and reports on indoor radon measurement studies conducted across the country with the aim of establishing a baseline data for the current study, which focuses on the development of a national radon survey and radon mapping strategy in SA. Moreover, findings made in this desktop review will inform the development of the national indoor radon database and the establishment of a regulatory framework for radon in dwellings and other buildings with high occupancy by members of the public.

Keywords: *Radon gas, Radon in dwellings, NORM, Geology, Uranium, Mining impact, South Africa, Regulatory Framework.*

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OS7.4 (T7.1-0633)

Application of the Closed Compartment Model (CCM) for a radon source characterization

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Radon is a radioactive gas considered as a carcinogenic element by the World Health Organization (WHO)[1] due to two of its descendants, ^{218}Po and ^{214}Po, which are alpha emitters. It is naturally occurred by uranium and radium decay and it can be found in rocks and soils, groundwater or building materials. Owing to its gaseous nature radon escapes from its sources from where it is transported into the atmosphere during its 3.82-day half-life.

In enclosed locations such as industries, public buildings or dwellings radon (^{222}Rn) tends to accumulate in the air, reaching high concentrations which are harmful to humans. It is important to control this levels of radon so, in this research, it is analyzed a mathematical model (Closed Compartment Model[2,3], CCM) to predict the concentration of radon in air from a solid source. The emission rate of the source and the maximum concentration that can be reached in air is also determined by the CCM.

For this purpose, measures have been carried out with two different systems. In both cases, the experimental devices used to measure ^{222}Rn in air are two high-density polyethylene sealed tanks, impermeable to radon: one of a 120 L volume and the other, 30 L volume. As a radon source two pitchblende stones, with different activities (Bq·kg^{-1}), have been placed inside the tank, just in the bottom. Above each one, there is a metal holder where the detectors are placed in order to measure radon in air: RadonScout Plus[4], CorentiumPro[5] and RadonScout PMT[6], all of them, measuring continuously.

During the exposure period, the deposit is hermetically sealed in order to keep the different atmospheric variables constant (pressure, temperature and relative humidity).

From the results obtained in this research it has been verified that the experimental data of radon in air measured inside the two tanks, with different pitchblende stones, follows the exponential equation developed by the Closed Compartment Model with an $R^2 = 0.99$. It has also been possible to obtain the emission rate of both radon sources and the maximum concentration of radon in air that would be reached in the system.

These results suggest that this mathematical model can be applied to different air volumes as well as different radon sources. In this way, the CCM could be applied for real installations or dwellings where the concentration of radon in air needs to be predicted in order to reduce the risks to the population.

Keywords: Radon in air, Closed Compartment Model, RadonScout Plus, CorentiumProu, RadonScout PMT

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OS7.4 (T7.B-0478)

Communication strategy for stakeholder of the real estate market in the context of Radon at the Czech Republic

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Communication strategy for stakeholder of the real estate market in the context of Radon at the Czech Republic

The State Office for Nuclear Safety recently noted several lawsuits concerning the sale and purchase of family houses. The lawsuits were conducted based on legal actions filed by the buyers who demanded a reduction of the purchase price or withdrawal from the contracts. The main argument of the actions, as indicated in the judgments, was a hidden defect reported by the buyers - the volume activity of radon about which the buyers had not informed.

The courts granted the actions and issued judgements in which they recognised the actions as justified and identified the presence of volume activity of radon and gamma dose rate as a hidden defect, based measurements and opinions provided by experts appointed by the courts.

The State Office for Nuclear Safety (SONS), based on its findings and the need to inform the stakeholder, prepared a communication strategy for the process of sale and purchase of real estates market about volume activity of radon, in 2019.

The strategy has several steps like selected the most visited real estate website and share in website short banner about radon as a hidden defect (goal is to attract visitors) with click to open more information. The easy banner will contain important information and offer for free an annual measurement by a passive detector, contact to experts. Other step wills information webinar about this topic with three experts of radon. The webinar is shared online and to streamed to YouTube permanently.

I want to present this communication strategy, results, weak points and experience.

PS1 (T1.1-0034)

ADAPTATION OF TWO PATIENTS WITH ACUTE RADIATION SICKNESS AND ACUTE LEUKEMIA AFFECTED IN THE ACCIDENT AT CHERNOBYL NPP

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Objective: The aim of the work is to assess the psychophysiological adaptation of two patients with severe arthritis and ARS of moderate severity and acute leukemia affected in the Chernobyl accident according to the characteristics of their individual mental adaptation and stages of the hematological syndrome (22 and 30 years of observation).

Material and Methods: Two patients were examined, former employees of the Chernobyl emergency shift. One of them acted as an engineer (Yu.A.P., 1961), the second - the deputy head of the turbine shop (D.R.I, 1950). They are 04/26/1986 were subjected to acute external relatively uniform gamma-beta irradiation due to the Chernobyl accident. Yu.A.P. he received Acute Radiation Sickness of the III degree of severity, multiple radiation burns of the I – II degree (40%) and the III – IV degree (15%) of the body surface. D.R.I. - ARS II severity and oropharyngeal syndrome I degree. The radiation dose, according to the cytogenetic study, was 4.3 Gy (Yu.A.P.) and 3.4 Gy (D.R.I.). Education in both higher. Disabled persons of group II with 100% disability. Married, in families of 1 and 2 children, children were born before the Chernobyl accident, were at the age of 2 and 7-10 years. Yu.A.P. in January 2007 he was diagnosed with chronic myeloid leukemia PH-positive in the acceleration phase. At D.R.I. - in June 2016 AML, transformation of myelodysplastic syndrome (MDS). Despite ongoing therapy, the death of Yu.A.P. came in November 2008, in D.R.I. in March 2017. Psychophysiological examination was conducted in 1999, 2001, 2016, i.e. 13, 15 and 30 years after the Chernobyl accident. Psychophysiological research was carried out using the method of MMPI, Cattell, Raven test, sensorimotor reactions of the automated program-methodical complex "Expert", designed to study the personality characteristics of a person, cognitive and intellectual personality traits 22 and 30 years after the radiation accident in dynamics.

Results: A clinical and psychophysiological assessment of the personality and the actual mental state made it possible to determine the following two types of impairment of psychophysiological adaptation: the type of asthenic depression and the demonstrative type of impairment of psychophysiological adaptation.

Conclusion: Psychophysiological assessment of personality and actual mental state determined individual personality traits in two patients in the form of asthenic depression with depressive-hypochondriac tendencies in one and a demonstrative type of violation of psychophysiological adaptation with pronounced hypochondria and a tendency to anxiety-depressive type of behavior in the other who had ARS, local radiation injuries of severe and extremely severe and acute leukemia after 22 and 30 years after the Chernobyl NPP accident.

Key words: acute radiation sickness, local radiation injuries, acute leukemia

PS1 (T1.1-0271)

Low-dose X-ray Irradiation does not cause Detrimental Effects in the Progeny of Irradiated Mesenchymal Stem Cells

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Current uncertainties in measuring the health risks associated with exposures to low-dose radiation present a major scientific problem affecting various areas of human activities, such as radiological protection of the pubic and nuclear industry workers, safety of patients exposed to low-dose diagnostic procedures, space exploration and others. A special interest is given to the studies of the delayed low-dose radiation effects induced in the stem cells. Repair of low-dose induced DNA double-strand breaks (DSBs) has been hypothesized to be inefficient, potentially causing various malfunctions in the progeny of the irradiated stem cells, such as accelerated cellular senescence or malignant transformation. The aim of the present work was to study the γH2AX and pATM foci (markers of DNA DSBs), cellular senescence and proliferation capacity changes in the progeny (up to 12 post-irradiation passages) of cultured human mesenchymal stem cells (MSCs) exposed to X-ray radiation at low- (40 and 80 mGy) and intermediate- (1000 and 2000 mGy) doses. Immunocytochemical methods and the senescence-associated β-galactosidase (SA-β-gal) staining were used for the analysis. We show that the progeny of cells exposed to low doses did not differ from the progeny of control non-irradiated cells at any time-point for all measured end-points. This was in contrast to cells treated with intermediate X-ray doses that showed changes that can be interpreted as detrimental. Thus, we recorded increasingly higher γH2AX and pATM foci rates, suppression of proliferation and promotion of cellular senescence as the passage number increased in the progeny intermediate dose irradiated cells. Taken together, our results show that low-dose X-ray radiation do not lead to the delayed radiation effects associated with the cellular senescence and genome instability in the progeny of irradiated MSCs.

Keywords: *Mesenchymal stem cells, Low-dose radiation effects, Delayed radiation effects*

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PS1 (T1.1-0311)

Effect of X-ray irradiation on dedifferentiation-mediated tumorigenesis in *C. elegans*

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How a committed cell can be reverted to an undifferentiated state is a central question in stem cell and tumor biology. This process, called dedifferentiation, is important for replacing stem cells as they get damaged, but it is also associated with the generation of tumor-initiating cells. Although cellular dedifferentiation (also tumorigenesis) has been observed in tissue culture cells and in organisms, its cellular behavior against X-ray irradiation remains poorly understood.

Using the nematode *C. elegans* as a model system, we recently demonstrated that X-ray (80 Gy) irradiation dramatically inhibits the proliferation activity of both normal germline stem cells and germline tumor cells (1). In this study, we next studied the effect of X-irradiation on dedifferentiation-mediated tumorigenesis using an exceptional *puf-8; lip-1* mutant worm. The *puf-8; lip-1* double mutant worms produce sperm continuously at the permissive temperature (20°C) (2), but they form germline tumors via the regression of differentiating spermatocytes into mitotically dividing cells (dedifferentiation) at the restrictive temperature (25°C) (3). Using this mutant worm, we tested the effect of X-ray on the initiation of dedifferentiation-mediated germline tumorigenesis at 20°C. Briefly, *puf-8; lip-1* mutant worms at the L4 stage were exposed to a 320 kV X-rays (0 and 80 Gy) and incubated for 2 days at 20°C. The gonads were then separated and stained with EdU-labeling kit (a proliferative cell marker) and DAPI (a DNA marker) (Fig. 1A,B). Our results showed that the control group (0 Gy) did not initiate the dedifferentiation-mediated germline tumors at 20°C (Fig 1A,C). However, X-ray exposed group (80 Gy) dramatically initiated the formation of dedifferentiation-mediated germline tumorigenesis at 20°C (Fig. 1B,C). Our further genetic results indicate that X-irradiation is likely to arrest the meiotic division of spermatocytes and initiates the formation of germline tumors via the regression of the arrested spermatocytes into mitotically dividing cells.

Together, our novel findings suggest that X-ray irradiation can suppress the proliferation of normal stem cells and aggressive tumor cells, but it can also initiate the formation of dedifferentiation-mediated tumorigenesis, depending on genetic context.

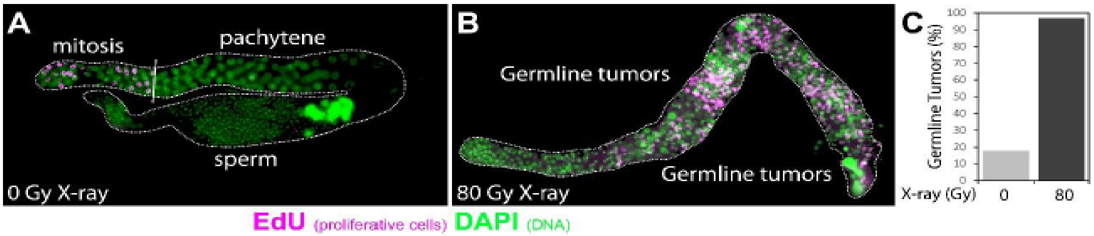

Figure 1. X-ray irradiation initiates germline tumors via dedifferentiation in *puf-8; lip-1* mutant germline.

Keywords: Dedifferentiation, Germline Tumor, X-ray, C. elegans

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PS1 (T1.1-0806)

Maternal preconception radiation exposure: solid cancers in offspring

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Preconceptional exposure of parents is considered as a potential carcinogenic factor for offspring [1]. Objective: comparative analysis of the incidence of solid cancers in the offspring of female workers of the Production Association "Mayak", subjected to prolonged preconceptional radiation exposure in the workplace. We carried out retrospective analysis in the group of offspring born in 1949-2004, including 4303 children from 2691 mothers. We followed up the vital status and cancer incidence of offspring until December 31, 2014. The number of person – years of follow-up in the cohort was 138072.3. We carried out the analysis of the structure of solid cancers, "crude" and standardized by age and sex incidence rates. Standardized incidence ratio (SIR) for solid cancers was calculated. The comparative analysis of cancer incidence was carried out by the method of indirect standardization; we applied regional and national standards as well as Poisson regression module AMFIT of EPICURE software [2].

Results: The range of accumulated preconceptual doses from external gamma-irradiation of the mothers was 0.01-5062.6 mGy with an average dose to the whole body 387.1 mGy, the dose to the gonads – 299.7 mGy. One third of mothers had accumulated preconceptual doses of internal alpha irradiation of the gonads with a maximum dose on the ovaries of 102.5 mGy. For 65 years (from 1949 to 2014) 116 cases of solid malignant neoplasms took place in the cohort of offspring of female workers of Mayak PA. The early onset of the oncological process occurred - the average age of the offspring was 46.6 years. The structure of solid cancers among the offspring of exposed mothers differed from national and regional statistics in the form of excess of the specific weight of the malignant brain tumors among the offspring of both sexes and thyroid cancer among women. There was a significant increase in thyroid cancer for women over 15 years in comparison with national data - SIR 2.33 (1.00-4.51). Malignant brain tumors in the offspring of both sexes older than 15 years significantly exceeded the national standard: among men – SIR 4.13 (1.64-8.37), among women – SIR 3.58 (1.11-8.32) and the regional standard among men older than 15 years – SIR 3.51 (1.39-7.10).

Conclusion: The multifactorial nature of solid cancers does not allow to unambiguously associate the results with the radiation exposure parents. We can only assume that maternal preconceptual radiation exposure plays a certain role in the carcinogenesis in offspring, which requires continued monitoring of the cohort of the Mayak Production staff offspring.

***Keywords**: Solid cancers, Maternal radiation exposure, Offspring*

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PS1 (T1.1-0844)

Characteristics of Human Lung Adenocarcinoma Cells that Survived Multiple Fractions of Ionizing Radiation

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Radiation therapy is one of the main methods of treating patients with non-small cell lung cancer (NSCLC). However, the resistance of tumor cells to exposure remains the main factor limiting the successful therapeutic outcome. The study of molecular-cellular mechanisms of increased resistance of NSCLC to ionizing radiation (IR) exposure is necessary to design approaches to reduce it and prevent tumor recurrence and metastases. Using fractionated irradiation of lung adenocarcinoma cells (A549), we obtained a population which survived total dose of 60 Gy and named it as A549IR. Further characterization of these cells showed multiple alterations compared to parental A549 cells. It was shown that the proliferative activity of A549IR cells was statistically significantly lower compared to control cells ($p < 0.05$). The number of γH2AX foci (marker of DNA double-strand breaks) 30 min after exposure to additional dose of 2 Gy was significantly higher in A549IR cells compared to unirradiated control ($p < 0.05$). Despite the fact that high number of γH2AX foci in A549IR cells may be linked to ineffective DNA DSBs repair at early time points, these cells demonstrated the decrease in the amount of γH2AX foci during 4-8 h after irradiation compared to parental cells. However the statistical significant difference was observed only 8 h after irradiation. The number of residual γH2AX foci 24 hours after irradiation to doses of 2, 4, and 6 Gy did not differ between parental and A549IR cells. Nevertheless, the number of colocolized γH2AX /53BP1 foci which represent the "true" DNA DSBs was significantly lower in A549IR cells exposed to 2 and 6 Gy compared to parental A549 cells. It is known that loss of p53 allows genetic instability as well as expansion or acquisition of stem cell features during carcinogenesis. The levels of p53 expression in A549IR cells was significantly lower after X-rays exposure to doses of 2 and 6 Gy ($p = 0.005$ and $p = 0.006$, respectively). At the same time, fractionated X-rays exposure in total doses of 2, 4 and 6 Gy led to an increase in the fraction of CSCs (side population, SP) in the A549IR cells ($p = 0.0002$ and $p = 0.0018$, respectively). In general, the results indicate that established A549IR cells demonstrate more effective DNA DSBs repair and decreased p53 activity which can promote tumor repopulation due to CSCs. The study was supported by the Russian Science Foundation (RSF) project # 19-74-10096.

PS1 (T1.1-0959)

Contribution of Childhood Radon Exposure to Lung Cancer Incidence among Youths and Young Adults: A Population-Based Study in Canada

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Radon is a naturally occurring radioactive gas generated by the decay of uranium-bearing minerals in rocks and soils. Since radon is a gas, it can move freely through the soil, enabling it to escape into the atmosphere or seep into homes and buildings. In outdoors, it gets diluted and does not pose a health risk. However, in confined spaces, i.e. indoors, radon can accumulate to relatively high levels and become a health hazard when inhaled. Exposure to indoor radon has been determined to be the second leading cause of lung cancer after tobacco smoking (1).

Canadian statistics show that most Canadians spend about 70% of their time indoors at home, 20% indoors away from home and 10% in outdoors. Due to relatively higher radon concentration in residential homes and longer time spent indoors at home, the exposure at home contributes to 90% of the radon-induced lung cancer risk(2). Even though lung cancer has not been observed in children, age-specific risk calculations indicated a shape increase in lung cancer incidence among youths and young adults based on linear relative risk model(2). Here, we conduct a population-based study to investigate the potential contribution of childhood radon exposure to lung cancer incidence among youths and young adults.

Following the launch of the National Radon Program in 2007, Health Canada completed a long-term radon survey in 33 census metropolitan areas (CMAs), which covers about 70% of the Canadian population(3). We used this data, together with available lung cancer incidence rates among youths and young adults (15 - 45 years of age at time of diagnosis) in 28 out of 33 CMAs in the past decade (2007-2016)(4), and tried to link the city-level average radon concentrations to the lung cancer incidence rates in those major Canadian cities. Findings on the contribution of childhood radon exposure to lung cancer incidence among youths and young adults are reported here.

Keywords: *Radon, Lung Cancer, Children, Young Adults*

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PS1 (T1.1-1005)

Cancer Risk among South Korean Medical Radiation Workers, 1999-2016

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Medical radiation workers are the largest group of workers with occupational radiation exposures and their number is rapidly increasing worldwide. In addition, the nature of radiation exposures in the workers is generally qualitatively similar to those received by the general population. However, little is known about the impact of job-related factors including protracted low-dose ionizing radiation exposure on cancer risk in adulthood. This study examined an overall evaluation of cancer risk among medical radiation workers as well as radiation effects on cancer rates. Data on all South Korean diagnostic medical radiation workers enrolled at the national dose registry between 1996 and 2011 were merged with the death and cancer incidence data until December 31, 2016. Occupational radiation doses were estimated for each worker based on individual badge measurements and work history. Standardized incidence ratios (SIRs) were used to compare the observed cancer incidence rates in this population to those for the general population while internal comparisons were used to estimate relative risks (RRs) for job-related factors. Excess relative risks (ERRs) for all solid cancer combined were calculated to quantify the radiation dose-response relationship using Poisson regression models adjusted for sex, attained age, and birth year. A total of 3,013 first primary cancer cases (2,860 solid cancer, 153 non-solid cancers) were identified among 93,870 diagnostic medical radiation workers. The mean cumulative badge dose among the cohort was 7.20 mSv which ranged from the minimum detectable level to 603.6 mSv. Compared to the general population, the rates of all solid cancers combined were significantly decreased in men (SIR 0.88, 95% confidence interval [CI] 0.84 to 0.92) whereas significantly elevated in women (SIR 1.12, 95% CI 1.05 to 1.18). The SIR for non-solid cancers was significantly increased in men (SIR 1.22, 95% CI 1.03 to 1.46). However, RRs for solid and non-solid cancers showed similar patterns by job titles, age groups, years of birth and entry to the registry, types of medical facilities, and cumulative occupational radiation doses among diagnostic medical radiation workers. There were no indications of a significant radiation dose effects on solid and non-solid cancer rates for either men or women (ERR/100 mGy <0.01 lagged 10 years for solid cancers and 5 years for non-solid cancers). The ERRs did not vary across job titles and other occupational factors. Occupational factors and radiation doses did not significantly associate with cancer incidence among South Korean diagnostic medical radiation workers although the cancer rates were significantly different compared to the general population. However, our null findings should be interpreted with caution because of uncertainties in dosimetry and model specifications. Further follow-up is required to better understand the effects of occupational radiation exposure on cancer incidence.

PS1 (T1.1-1019)

Additive effect of ionizing radiation and sodium-acetylsalicylate increases leukemia cells death. Implications for radioprotection of patients

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Blast crisis is a terminal phase of chronic myelogenous leukemia (CML), characterized by a rapid expansion of a population of myeloid cells and by an average survival rate of two years.
CML is associated with a reciprocal translocation between chromosome 9 and 22. This translocation generates BCR/ABL fusion gene which encodes different chimeric proteins with strong tyrosine kinase activity generated by several alternatives of fusion genes.
Since its clinical introduction over a decade ago, first and second generation of tyrosine kinase inhibitors (TKIs), have revolutionized the treatment of chronic phase of CML but TKIs resistance remains a major impediment to successful treatment for approximately 15- 20% of patients.
In these cases, allogenic bone marrow transplantation (BMT) may be required after total body irradiation (TBI) with doses ranging from 10 to 16 Gy with different fraction schemes. TBI has become a widely used conditioning regimen for BMT or peripheral blood stem cell transplantation in patients with hematological malignancies, multiple myeloma, Hodgkin´s or non-Hodgkin´s lymphomas. TBI is the most effective immunosuppressive conditioning agent because it is able to gain access to areas where effective chemotherapy levels cannot be reached.
In the latest years, TBI combined with chemotherapy or radiomimetic, has evolved but is still responsible for several cases of acute and late toxicity in lungs, heart, liver and kidneys.
Nonsteroidal anti-inflammatory agents are known for their strong analgesic, antipyretic and anti-inflammatory activity and for decreasing the survival of different human cancer cells.
The objective of our work was to analyze *in vitro* if there is an additive effect between sodium-acetylsalicylate (Na-Sal) and ionizing radiation to increase the death of leukemia cells in blast crisis.
Eritroleukemia cells (K562 cell line) were cultured in presence of 0, 2 and 4 mM of Na-Sal. After 72 hours they were gamma irradiated with 3 Gy at the dose rate of 200 mGy/min. Immediately after irradiation 1000 cells of each concentration irradiated and non-irradiated, were seeded in semisolid medium in 25 cm^2 tissue culture flask for cloning survival determination.
After 20 days, cloning survival of cells treated with Na-Sal and Na-Sal plus 3 Gy gamma irradiated was determined. (Table 1)

Table 1: percentage of cloning survivals at different Na-Sal concentrations and radiation

Na-Sal concentration (mM)	Cloning survival %	
	0 Gy	3 Gy
0	100*	45
2	100	22
4	16,5	0,5

*This value was considered to obtain the percentage of different cloning survivals

In the present study, we have examined the additive effect of Na-Sal and gamma radiation on clonogenic survival of K562 cell line. Our results show that Na-Sal and ionizing radiation contribute synergistically to cell death of eritroleukemia cells in blast crisis. For this reason, Na-Sal, it becomes a promising potential therapeutic tool for treatment and could be a novel candidate to be used in combination with radiation, to decrease the necessary dose to obtain total neoplastic cell death prior BMT. Also, lowering necessary dose to kill cells could reduce adverse events of radiation in critical organs, contributing to radiation protections of patients, particularly for young and children.

Keywords: *eritroleukemia, gamma-radiation, sodium-acetilsalicylate.*

PS1 (T1.1-1181)

Cohort profile of the Japanese epidemiological study on low-dose radiation effects (J-EPISODE)

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The Institute of Radiation Epidemiology of the Radiation Effects Association has examined radiation epidemiological study among Japanese nuclear workers since 1990. A new study was designed with a background of that incidence data were needed in addition to mortality data and so on. The new study has been conducted during 2015 to 2019 to obtain new informed consent and information of confounding factors by lifestyle questionnaire survey. For those expressed agreement to informed consent were requested to answer the lifestyle questionnaire simultaneously. The questionnaire was the self-administered and included questions about lifestyle such as smoking and occupation, etc. The documents were distributed in two ways. The first was distributing by mail to those included in the previous cohort whose their name, addresses and dose records were identified The second was distributing to those currently working through the organization of nuclear facilities such as nuclear power plants, research institute, and fuel processing companies. The worker who replied in the second way, data linkage with database which is maintained by Radiation Dose Registration Center by using their name, date of birth and address have performed to link their dose records. Based on these surveys, a new cohort which was comprised by 77,993 male workers was established. The mean cumulative dose was 15.4 mSv and the mean age was 59.4 years at the end of March, 2019. The workers who had exposed less than 5 mSv or who were over 60 years old occupied half and over. Duration of work, type of employer, job category, years of education, smoking status and body mass index showed correlation with cumulative dose. Alcohol consumption did not show the correlation. These results suggest that the estimated excess relative risks per sievert will reduce by adjustment for them as same with the previous analysis[1]. The characteristics of new cohort denoted that adjustment for lifestyle or socioeconomic status should be needed in future analysis.

Keywords: *Epidemiology, Cohort study, Nuclear worker*
ACKNOWLEDGMENTS: This work was funded by Nuclear Regulation Authority, Japan.

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PS1 (T1.1-1248)

Papillary thyroid carcinoma genetic susceptibility factors in children exposed to radioiodine from Chernobyl fallout

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Background: Statistically significant associations between papillary thyroid cancer (PTC) and specific common single nucleotide polymorphisms (SNP) in *MGMT* (rs2296675), promotor region of *FOXE1* (rs1867277) and *ATM* (rs1801516) genes were reported in Belarusian children exposed to Iodine-131 (^{131}I) after the Chernobyl nuclear power plant accident. We further pursued genetic determinants associated with individual susceptibility to radiation-related thyroid carcinoma (TC) within the framework of the EPITHYR consortium.

Methods: We investigated the associations between PTC risk and SNPs in candidate DNA repair genes, ATM and FOXE1 genes in 343 individuals who were <15 years old at the time of the Chernobyl accident, including 66 histologically verified PTC cases diagnosed between 1992 and 1998, and 277 population-based controls, genotyped using the OncoArray (Illumina). We used an arithmetic mean of 1,000 individual stochastic thyroid doses due to ^{131}I intake calculated using Monte-Carlo simulation method as radiation dose exposure. We evaluated the associations between 122 SNPs and PTC using logistic regression models assuming a log-additive allelic inheritance models and taking into account population stratification, and effects of age, sex and thyroid dose. We accounted for multiple testing by applying the Benjamini & Hochberg correction.

Results: We found some suggestive associations ($P < 0.05$) between PTC risk and polymorphism in the following genes: *MGMT* (rs553371, rs4751107, rs12251275, rs477692, rs7078706, rs1008982), *NBN* (rs1063053), *XRCC1* (rs1799782), XRCC5 (rs3835, rs3821107, rs6729441, rs1051685, rs7583902) involved in DNA repair, and *ATM* (rs1801516) involved in ATM signalling pathway. However, after adjustment for multiple testing, none of these SNPs remained statistically significant. Analyses on other possible pathways, as well as on potential gene-radiation interaction are ongoing.

PS1 (T1.2-0050)

Adhesion molecules expression: Beta 1 integrin and ICAM-1 as potential markers of cutaneous radiation injuries induced by ionizing radiation overexposures

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During the inflammatory response, there is a process of leukocyte extravasation that involves the migration of these cells from the bloodstream towards target tissues through their adherence to vascular endothelium. Leukocyte extravasation is coordinated and regulated by the expression of a variety of glycoproteins implicated in cell-cell interactions. Cell adhesion molecules (CAMs) mediate interactions between blood cells and endothelial cells that can occur in all segments of the microvasculature as a response to inflammation under certain conditions such as overexposure to ionizing radiation.
On the other hand, the beta integrin family of proteins interacts with the associated ligand (intercellular adhesion molecules) in the vascular endothelium. This transient binding results in further leukocyte activation and subsequent firm adhesion and transendothelial migration into sites of inflammation.
The present study examines the expression of two of these proteins: Beta 1 integrin and ICAM-1, using flow cytometry and immunohistochemical techniques, in blood samples and biopsies of patients overexposed to ionizing radiation.
This work shows the correlation between the expression of Beta 1 integrin in lymphocytes from blood samples and the expression of its associated ligand in endothelium ICAM-1 and the possible role of this interaction between these molecules in the initial phases of infiltration into the tissue affected by radiation exposure.

Keywords*: Integrin, ICAM, Overexposure*

PS1 (T1.2-0276)

Study of the effects of low doses of ionizing radiation on human stem cells

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During life a person is exposed to low radiation doses: background radiation, within the framework of medical diagnostics and treatment, from radioactive waste dumps, during professional activities and in air travel. Despite the increasing number of sources of low-dose radiation the effect of low radiation doses on human stem cells remains unstudied to date, disturbances in the vital processes of which directly affect the body as a whole [1]. The aim of the study was to study the effect of low X-ray doses on human mesenchymal stem cells (MSCs) in long-term cultivation *in vitro*.

It was shown that the proliferative activity (PA) of MSCs of the mucosal gum tissue which were irradiated at dose 50 mGy is comparable with the control group in long-term cultivation while doses of 100 and 250 mGy showed a decrease of PA. Also non-irradiated MSCs showed a significant decrease of the PA during cultivation in a conditioned medium from cells that received dose of 1000 mGy and an increase of PA during cultivation in a conditioned medium of cells that received doses of 50, 100 and 250 mGy. The cells were previously irradiated at dose 250 mGy showed adaptive response during cultivation in conditioned medium from cells that received dose of 1000 mGy.

So the current contradiction of the results of studies of the effects of low radiation doses on living systems does not answer the question "Do the effects of low doses mainly correspond to the threshold or non-threshold concepts of the effects of radiation on the body?".

The assessment of the effects of low radiation doses was focused on the "bystander effect" [2] in the presented study. It was noted after adding conditioned media from irradiated cells to previously irradiated and non-irradiated MSCs. The inhibitory effect of conditioned media from cells that received a sublethal dose of radiation was noted in the same way as in the work of Mazersil et al. Suggesting that this effect is positive and is aimed at eliminating already damaged cells from the population [3]. In comparison with the effect of conditioned media from cells that received low doses on PA, it can be definitely said that the "bystander effects" for low and high doses are different and their biological meaning requires further study.

The phenomenon of adaptive response was shown after addition conditioned media from cells irradiated at dose 1000 mGy to pre-irradiated MSCs received a dose of 250 mGy. Cells showed higher levels of PA compared with other irradiated groups. The obtained result casts doubt on the assumption made by Mazersil et al., and also leads to the conclusion that the effects of low doses can be positive.

Thus, the results of study mainly correspond to the threshold nonlinear concept, according to which the effect is not proportional to the received radiation dose. The cytokine profile of the conditioned media of irradiated MSCs will be studied in continuation of the study.

Keywords: human stem cells, low doses, X-ray.

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PS1 (T1.2-0050)

Adhesion molecules expression: Beta 1 integrin and ICAM-1 as potential markers of cutaneous radiation injuries induced by ionizing radiation overexposures

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During the inflammatory response, there is a process of leukocyte extravasation that involves the migration of these cells from the bloodstream towards target tissues through their adherence to vascular endothelium. Leukocyte extravasation is coordinated and regulated by the expression of a variety of glycoproteins implicated in cell-cell interactions. Cell adhesion molecules (CAMs) mediate interactions between blood cells and endothelial cells that can occur in all segments of the microvasculature as a response to inflammation under certain conditions such as overexposure to ionizing radiation.
On the other hand, the beta integrin family of proteins interacts with the associated ligand (intercellular adhesion molecules) in the vascular endothelium. This transient binding results in further leukocyte activation and subsequent firm adhesion and transendothelial migration into sites of inflammation.
The present study examines the expression of two of these proteins: Beta 1 integrin and ICAM-1, using flow cytometry and immunohistochemical techniques, in blood samples and biopsies of patients overexposed to ionizing radiation.
This work shows the correlation between the expression of Beta 1 integrin in lymphocytes from blood samples and the expression of its associated ligand in endothelium ICAM-1 and the possible role of this interaction between these molecules in the initial phases of infiltration into the tissue affected by radiation exposure.

***Keywords**: Integrin, ICAM, Overexposure*

PS1 (T1.2-0588)

Novel Therapy of Scabies's Disease

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Background: Scabies is a common disease that is caused by *Sarcoptes scabiei var. hominis*. This disease is usually characterized by skin lesions which form straight or winding cuniculus coloured white or grey. These lesions generally located on the thin stratum corneum. It often accompanied by itching sensation at night and quickly spreads in dense environments. Scabies is a skin disease that is endemic in tropical and subtropical climates. In Indonesia, there are 14,798 Islamic boarding schools with a high prevalence of scabies.[1] Treatment using permethrin 5% is felt to be less effective because if it not done simultaneously and massively, recurrence will occur immediately. The scabies disease certainly makes the students frustrated and decreased educational value.[2]

Materials and Methods: The utterly randomized design used in this experiment. This research conducted at some Islamic boarding houses in East Java, Indonesia. There are 189 samples were obtained, ranging in age from 13-24 years. These samples randomized, divided into three groups. The scabies patients treated by permethrin 5% as the positive control. The negative control was scabies patients without any treatment. And the last group were scabies patients treated by ultrasonic sound (70.000 Hz). The duration of ultrasonic sound was 3 hours each night. The severity before and after treatment were measured using the Scoring System of Severity in Scabies. The degree of decrease in scabies skin lesion using ultrasonic within 3 days can be seen in Figure 1.

Figure 1. The lesion repair using ultrasonic sound

Results: In ultrasonic treatment, there is a drastic reduction in the degree of disease. In therapy using permethrin 5%, the decline is not so significant. Meanwhile, without treatment, the degree of illness is increased. Based on the results of the Kruskal Wallis test, the calculated chi-square value is higher than the chi-square table (61,994> 5,991). The p-value is smaller than α (0,000 <0.050), the H0 decision rejected, so it could conclude that there is a significant difference in the average decrease in the degree of disease between treatments.

Conclusions: The ultrasonic sound can consider as a novel therapy for scabies disease because the efficacy in time and also the repair skin lesion.

***Keywords**: Ultrasonic, Scabies, Novel Therapy*

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PS1 (T1.2-0876)

Oligophrenia prevalence among persons exposed antenatally

Shalaginov S.A., Mikryukova L.D., Akleyev A.V.

As it was shown in Hiroshima and Nagasaki, one can expect an increase in child's birth with mental retardation after exposure of forming brain. It is also well-known that the reasons causing moderate and severe mental retardation can be various. A significant part of severe oligophrenia should be the cases resulting from exogeneous embryotoxic factors. On the other hand, it would be logical to assume that a substantial part of mild oligophrenia cases were individuals with deprivation forms of oligophrenia as well as individuals with mental retardation resulting from high population inbreeding because of a limited choice of marriage partners.

Registry data for individuals with undifferentiated oligophrenia as well as dosimetry registry data taken from the Urals Research Centre for Radiation Medicine (Chelyabinsk, Russia) were used in this work. The data were collected in the period 1991-1995. Individuals born in the period 1950-1960 in the Techa riverside villages and still residing as of the end of 1995 in the territory of the 5 radioactively contaminated administrative raions of Chelyabinsk Oblast make up a group of persons exposed antenatally. Population control was formed of unexposed individuals residing in the adjacent "clean" territories.

Table 1. Prevalence, %, severe (imbecility, idiocy) and mild (debility) oligophrenia in the Techa River persons exposed antenatally

Dose groups, Sv	Offspring number	Imbecile and idiot number	Number of mild mentally retarded persons	Severe oligophrenia prevalence	Debility prevalence
< 0.001	479	1	9	0.21	1.88
0.001-0.0099	1,285	4	20	0.31	1.56
0.01-0.099	667	2	9	0.30	1.35
> 0.1	291	2	4	0.69	1.37
Total exposed antenatally	2,722	9	42	0.33	1.54
Population control	30,938	52	343	0.17	1.11

One can note (table 1) a tendency of an increasing birth of children with distinct forms of mental retardation as a dose for fetus and embryo increases, as well as inverse relationship of dose for mild oligophrenia

PS1 (T1.2-0884)

Down's syndrome in the offspring of the Techa River residents exposed antenatally

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Based on the peculiarities of haematogenesis of men and women, it is necessary to approach asymmetrically the assessment of the effects in the offspring of exposed parents. The most important for the offspring of exposed mothers is the assessment of possibility of induced mutagenesis during the period of egg pool development (the first 3 months of a female embryo).

The Regional/Chelyabinsk Oblast registry of patients with Down's syndrome generated by the Urals Research Centre for Radiation Medicine in the period 1994-2005 is the basis for this work. The offspring of individuals exposed antenatally include persons born in 1977-1995 whose parents resided in the Techa riverside villages in the period 1950-1953. Persons residing in the adjacent Techa riverside settlements and born within the same period of time were selected as a control groupTable 1 shows that Down's syndrome frequency is 4.9 ‰ in the group with mother only exposed antenatally, and not a single case of Down's syndrome was registered in the groups with exposed fathers and exposed both parents

Table 1 – Down's syndrome frequency (‰) in different offspring groups of individuals exposed antenatally

Offspring groups	Number of children	Number of cases of Down's syndrome	Frequency of Down's syndrome, ‰
Both parents exposed	202	0	0
Mother exposed only	612	3	4.9
Total mother exposed antenatally	814	3	3.7
Father exposed only	730	0	0
Total both parents exposed	1,544	3	1.9
Control	18,176	21	1.2

Despite an extremely high indicator of Down's syndrome frequency in persons with mothers exposed antenatally (total group 1+2) – 3.7 ‰, a small sample size does not allow registering reliable differences when compared with the control group, frequency of Down's syndrome was1,2 ‰.

PS1 (T1.2-1076)

Evaluation of the Effects of Low-dose and High-dose Ionizing Radiation on *In Vivo* Models

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Living organisms are consistently exposed to low and very low levels of ionizing radiation during daily life. The understanding biological effects to low-dose radiation is an important issue in radiation protection field. Although various studies have investigated the biological effects of low-dose ionizing radiation (LDIR) using *in vivo* models, the implications of LDIR on living organisms remain unclear. The differences in observed biological effects on *in vivo* models may depend on doses and dose-rates. We have been conducting extensive researches on the biological effects of low-dose and high-dose ionizing radiation (HDIR) and its molecular mechanism in animal models using *Drosophila* and mice to compare differences between doses and dose-rates. Notably, *Drosophila* is an ideal model organism for aging studies due to its short lifespan and conserved molecular pathways. First, we analyzed the effects of LDIR treatment on the *Drosophila* lifespan. LDIR treatment extended the *Drosophila* lifespan, but did not alter developmental rates. In addition, the locomotive deterioration of aged flies was rescued by exposure to a 0.05 Gy dose of γ-irradiation. Because oxidative stress is a major cause of aging, we next investigated the effects of LDIR under oxidative stress conditions. The oxidative stress susceptibility of flies under hydrogen peroxide (H_2O_2)-induced oxidative stress conditions was evaluated at early post-irradiation time points. Interestingly, LDIR increased the survival rates and locomotive ability of flies under oxidative stress conditions and decreased mRNA levels of the pro-apoptotic factors, grim and reaper. Conversely, HDIR augmented the negative effects of oxidative stress. Next, to evaluate the effects of ionizing radiation in mice model, we examined hematological and biochemical parameters and histological changes after low- and high-dose of γ-irradiation. HDIR showed significant changes compared to control group, but LDIR didn't. These results suggest that the effects of LDIR may be interpreted differently than those of HDIR, since the biological response to ionizing radiation varies depending on doses and dose-rates. Finally, we'd like to contribute to public understanding about the effects of low-dose radiation by providing these scientific evidences.

***Keywords**: Ionizing radiation, Low-dose, Biological effect*

Poster Session T1

PS1 (T1.3-0421)

Radionuclide and Elemental Concentrations of some Food Beverages Consumed in Southwestern Nigeria

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Samples of thirteen of the bottled fruit juices produced and consumed in Nigeria were analyzed in the study. Concentrations of the following elements - Cd, Ca, Mg, Mn, As ,Al, Cl, Hg, Cu, Zn, K and Na were carried out using Instrumental Neutron Activation Analysis (INAA). The activity concentrations of naturally occurring radionuclides (NORMs) i.e. ^{226}Ra, ^{232}Th and ^{40}K was also determined using Gamma Spectrometry. The results of the study showed the distributions of the trace elements concentrations (in mg/L) to be <0.001 to 7.92±1.12, <0.01 to 0.43±0.06, 0.19±0.02 to11.45±1.72, 0.26±0.04 to1.43±0.21, 0.54±0.08 to 15.14±2.27, <0.01, <0.01, 0.88±0.13 to 6.27±0.94, 0.001 to <0.01, 0.97±0.03 to 22.06±0.14, 0.18±0.25 to 0.93±0.16 and 1.30 ± 0.19 to 12.5 ± 1.88 for Cu, V, Cl, Mn, Ca, Cd, Hg, K, As, Na, Al, and Mg, respectively. The ranges of activity concentrations (Bqkg^{-1}) of ^{226}Ra, ^{232}Th and ^{40}K in the food beverages were 0.12 - 1.72, 0.05 - 1.45 and 12.63 - 22.31, respectively. The estimated means of the annual effective dose and the internal radiation hazard indexes (H_{int}) were 0.005mSvy^{-1} and 0.072, respectively. Either elemental and radionuclides concentrations, in the food drinks studied do not present any radiological or chemical concerns when compared to international recommended and global average values.

PS1 (T1.3-0612)

The Frequency and Spectrum of Chromosomal Aberrations in Blood Lymphocytes of Healthy Occupational Employees Exposed to Chronic Irradiation

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In recent decades the ionizing radiation is increasingly infiltrate the everyday human life (using the radioactive isotopes and X-ray in medical diagnostic procedures, radiation therapy of cancer, etc.). The cytogenetic analysis of peripheral blood lymphocytes so far is the most common method of irradiation effects indication.
The main goal was to determine the spectrum and frequency of chromosomal aberrations in blood lymphocytes of healthy occupational employees exposed to different levels of ionizing radiation.
Material and methods. The incidence of chromosomal aberrations was analyzed in peripheral blood lymphocytes of occupationally exposed people (main group, n=1,308), so as individuals living in an area nearby the nuclear plants (control group, n=100). A standard cytogenetic analysis of blood lymphocytes (in at least 300 metaphases) has been carried out for all of the examined individuals. We analyzed only that types of chromosomal aberrations we were able to identify without karyotyping.
Results. The spectrum and the frequency of chromosomal aberrations in blood lymphocytes during external (γ-radiation), internal (α-radiation) and combined exposure were analyzed. We detected polyploids, rings, dicentrics, single, and paired chromatid fragments. Metaphases contained at least one chromosomal aberration were considered aberrant metaphases. The individuals exposed to external irradiation showed higher frequencies of all types of chromosomal aberrations than normal controls. The individuals exposed to combine irradiation showed higher frequencies of dicentrics than normal controls. The individuals exposed to internal irradiation showed equal frequencies of all types of chromosomal aberrations than normal controls. The individuals exposed to external irradiation showed different frequencies of all types of chromosomal aberrations than individuals exposed to combine irradiation. Chronic external irradiation appeared to be the main factor of induction of chromosome type aberrations.
A nonlinear dependence the dose of irradiation and frequency of chromosome aberrations was revealed. A statistically significant decrease of the frequency of the chromosome aberrations was established in employees exposed to irradiation at a dose range of 0–10 mSv compared to the control group. This agrees with the phenomenon of radiation hormesis. The threshold level of increasing the chromosome aberrations frequency depends on their type and implemented in the dose range of 10–200 mSv. The dose-effect curve has a plateau at doses ranging from 200 to 500 mSv for all types of chromosomal aberrations. The frequency of dicentrics was increasing in direct proportion with an external dose of more than 500 mSv, for other types of chromosome aberrations with an external dose of more than 800 mSv.
Conclusion. The strongest inducer of chromosomal aberrations is chronic external irradiation exposure, which at a dose of more than 10 mSv is accompanied by an increase in the frequency of chromosomal aberrations. The variability of the frequency of chromosomal aberrations at the same dose of irradiation is a manifestation of individual radiosensitivity in human.
Keywords: *healthy employees, chronic external irradiation exposure, chromosomal aberrations*

PS1 (T1.3-1158)

Cryptosporodium parvum, a novel model for the development of radio-protectors

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Background: *Cryptosporidium parvum* (Cp) is an obligate intracellular protozoon parasite that infects a wide range of vertebrates, including human and animal. Not only does Cp spread ubiquitously in our environment, it is also highly resistant to harsh environmental conditions and disinfectants. Previously, we have demonstrated that Cp exhibits the highest resistance to ionizing radiation among parasites [1,2]. In an attempt to explore the potential use of Cp for the development of radio-protectors, we investigated radio-protective effects of lysates prepared form Cp lysate.

Methods: Big (>10K) and small (<10K) fractions of Cp lysate were prepared using centrifugal filtration devices (Amicon). The radio-protective potential of Cp filtrate was evaluated in 2 normal cell lines viz. COS-7 (African green monkey kidney cells) and HaCat cells (Human keratinocyte cells).

Results: In MTT assay, in the presence of these lysates COS-7 and HaCat cells had higher survival rates as compared to control group upon irradiation with a dose of 2 Gy or 10 Gy, implying that Cp lysate protected the cells from radiation-induced damage. We further evaluated DNA damage (γH_2AX) and repair marker (Rad51) by immunofluorescence. Both the fractions prevented radiation-induced DNA damage (γH_2AX) in both the cell lines at 1 h as well as 24 h post irradiation (10 Gy). However, the small fraction exhibited better radio-protection in both the cell lines. The DNA repair marker also followed a similar trend in both the cell lines. We could see that Rad51 expression was lower in the lysates treatment group at both the time points. As we know that ROS is well recognized mediator of DNA damage, hence we measured ROS levle in both the cell lines by using H_2DCF-DA (by FACS) and CellRox green (by fluorescence microscopy). The small fraction scavenged ROS in both the cell lines more effectively as compared to big fraction.

Conclusion: Hydrophobic and aromatic amino acid residues in small fraction may be the key factors that determined the antioxidant activities of the proteins. In addition, ubiquitin-like protein might have conferred radioprotection.

Keywords: *Cryptosporidium parvum, radio-resistance, radio-protection*

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Poster Session T1

PS1 (T1.4-0418)

Assessment of Radionuclides Concentrations in Fertilized Soils from Farmlands in some parts of Abeokuta, Ogun State, Nigeria

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The activity concentrations of radionuclides in fertilized farm soils in some parts of Abeokuta, Southwest Nigeria were determined. The soil samples were randomly collected from 101 points in twenty farmlands comprising of 18 fertilized and 2 control sites. Absorbed dose rates were measured in-situ at each point using a Cesium Iodide-based dosimeter held at about 1m above the ground level. The in-situ measurements were compared with values of activity concentrations obtained from the soil samples collected and analyzed using a NaI-based gamma-spectrometer. Dose rates, annual effective doses, external hazard index and radium equivalent activities were calculated from the values of the activity concentrations. Pearson correlation test was used to compare the in-situ and the calculated absorbed dose rates. The range and mean values of the in-situ dose rates were (0.02 - 0.23) µSv/h and 0.062 µSv/h respectively. The ranges and means of measured activity concentrations (Bqkg^{-1}) were found to be (147.09±11.89 - 557.32±46.6) 319.51±32.39; (10.87±2.58 – 84.41±11.01) 25.08±6.28, and (16.50±6.02 - 158.69±17.78) 41.66±12.62 for ^{40}K, ^{226}Ra and ^{232}Th respectively. The mean values of calculated dose rates, annual effective doses, external radiological hazard indices and radium equivalent activities were 0.0295 µSv/h; 109.042 µSv/y; 0.243 and 90.08 Bq/Kg respectively. The mean in-situ dose rates were higher than the world weighted mean dose rate (of 0.04 µSv/h) by United Nations Scientific Committee on the Effects of Atomic Radiations. The mean in-situ dose rate and world average dose rate were higher than the mean calculated dose rate. The Pearson's correlation test (r = 0.924, N = 101, p = 0.000) indicates that there is high positive relationship between in-situ and calculated dose rates. The annual effective doses were lower than the recommended dose limit of 1mSv/y given by International Committee on Radiation Protection. There were indications that fertilizers applied to soils increased the concentrations of radionuclides in the study area.

PS1 (T1.4-0490)

Modeling of atmospheric dispersion and radiation dose for a hypothetical accident in Radioisotope production facility

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Atmospheric dispersion modeling and radiological safety analysis is performed for public outside radioisotope production facility (RPF) in case of hypothetical radioactive Iodine spilling and leakage from hot cell. Potential human error is expected and column that occupies Iodine may be broken causing it to spill on the hot cell floor. Ventilation system is dedicated to extract dispersed material through dedicated filters before gases expelled outside the facility. Two scenarios are presented in this paper, the first one is prediction the dispersion with good filtration from extract ventilation system, while the other with loss efficiency of filtration components. The spilled radioiodine is the source term, and the HotSpot 3.1, Health Physics code was created by LLNL [1] is used to provide Health Physics calculation tool for evaluating accidents involving radioactive materials to perform the atmospheric transport modeling which is then applied to calculate the total effective dose equivalent (TEDA) in different atmospheric stability classes, and how it would be distributed to human body as a function of downwind distance and radionuclide activity. The adopted methodology uses predominant site-specific meteorological data and dispersion modeling theories to analyze the impact of hypothetical release to the environment from the selected radionuclide and evaluate to what extent such a release may have radiological impact on public.

PS1 (T1.4-0736)

Radioactivity and the Environment: The UK wide research programme

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The RATE (Radioactivity and the Environment) programme was a UK based research programme designed to address key knowledge gaps associated with environmental radioactivity. In addition to addressing key knowledge gaps, a second key aim of is building UK capacity. RATE researched a wide range of topics examining different, interlinking parts of the environment: the subsurface environment, the near-surface environment and transfer and effects to non-human biota.

The research focussing on the subsurface environment characterised fractures in rock mass at a range of spatial scales, biogeochemical coupling of radionuclides and multiphase transport, and the role that microbes play in the availability of radionuclides in the subsurface environment. In the near-surface environment, the consequence of bio cycling (e.g. redox state) of radionuclides such as neptunium and technetium was investigated. Research was also conducted to research how anthropogenic complexants affect the solubility of radionuclides and the potential for plutonium to leach from cementitious waste forms. Closely associated with the near-surface environment is the role plant and fungi can play in the bioavailability of radionuclides. RATE research has shown that different root systems, the presence of arbuscular mycorrhizal fungi and root-induced changes can influence the amount of radionuclide uptake in plants.

Research was conducted to increase understanding and reduce the uncertainty associated with radionuclide transfer to wildlife and plants. A "REML approach" was developed to predict concentration ratios for wildlife which currently have no data. There was also a significant improvement in establishing transfer parameters for poorly studied organisms, radionuclides and ecosystems. Finally, effects were measured in a number of different wildlife and plants, focussing on previously less-studied organisms, including, fish, plants, crustaceans and insects.

This poster summarises some of the key findings of RATE. The collective research from the RATE programme will provide supporting information for the development of the UK's Geological Disposal Facility, further the understanding of contaminated legacy sites and potentials for bioremediation, improve dose-effect relationships for wildlife and provide evidence for the regulatory framework.
RATE was an £8.6m, a five-year research programme funded by Natural Environment Research Council, the Environment Agency and Radioactive Waste Management Ltd.

PS1 (T1.4-0907)

Paraguay's experience in the Re-export of Five Disused Co-60 sources. Radiological Protection Approach

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This paper presents a radiological protection approach from the perspective of Paraguay's experience in the re-export of five disused sources of Co-60 heads, which were used for cancer treatment (Teletherapy). The use of these Co-60 heads for radiotherapy purposes were replaced by tele-therapy technology with linear accelerators. These disused radioactive sources were distributed in three different facilities in the country. This paper presents the activities carried out in the management and conditioning for the safe transport of Co-60 sources, as well as nuclear materials (depleted uranium) that were used as shielding of the sources. In addition, all necessary documentation and requirements, and radiation protection approaches that were taken into account during the re-export procedure.

Keywords: *radiological protection1, shielding,*

PS1 (T1.4-0913)

Influence of a native grass on radionuclide transport in laboratory-scale soil columns

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This work seeks to quantify (1) the uptake of ^{99}Tc, ^{237}Np, ^{238}U, and ^{133}Cs in the grass species *Andropogon virginicus* from laboratory-scale soil columns and (2) the transport of these analytes through the columns in order to elucidate mechanisms that may affect this transport when plants are present. Scaling up from previous studies investigating influence of plant exudates on soil sorption as well as hydroponic plant uptake experiments, this work adds to the body of knowledge aimed at providing insight into potential factors contributing to migration of radionuclides in Savannah River Site (SRS; Aiken, South Carolina, United States) soil.

The ability of *A. virginicus* to alter analyte transport through a soil medium was studied on a macroscale with soil-plant columns. Columns consisted of semi-rigid 5 cm diameter clear plastic tubes filled with a 50:50 mixture of SRS soil:sand. Rhizon© pore water sampling tubes were inserted at depths of 7.6 cm and 17.8 cm from the top of the soil surface. These ports were used to introduce analytes into the column after plants were established as well as to sample the pore water at the completion of the experiment. Three groups of columns with six replicates each were constructed: (1) columns with plants and radionuclides, (2) columns without plants and with radionuclides, and (3) control columns with plants and without radionuclides. *A. virginicus* seedlings were transplanted into the columns from a germination mixture when they had between 3 and 5 leaves and were allowed acclimate in the columns for approximately 3 weeks before the analytes were introduced into the top port. Columns were housed in a growth chamber (25 °C, 14-hour light cycle) and irrigated from the surface every 2 to 3 days alternating between Hoagland nutrient solution and water; all column effluent was collected for analysis. Shoots were harvested four weeks after analyte introduction. The columns were then covered and stored at 5 °C until segmentation. To segment columns, the plastic was cut lengthwise and the soil was cut into 1 cm segments and retained for root recovery as well as soil digestion and analysis. Samples (aqueous, plant digestate, soil digestate) were analyzed with inductively coupled mass spectrometry (ICP-MS) and/or liquid scintillation counting (LSC) as appropriate.

Analysis of effluent and soil digestate samples indicates that plant presence retards downward transport of ^{99}Tc and ^{237}Np through the column with respect to amount of irrigation water (or solution) introduced and contributes to upward migration of these mobile radionuclides. The reasoning for this is multifold; it includes, in part, the evapotranspiration action of the plant, which greatly affects transport of water and mobile ions through the soil column, and it is influenced by plant uptake, particularly for ^{99}Tc as demonstrated from analysis of the plant tissues. Effects on the transport of ^{133}Cs and ^{238}U could not be discerned from aqueous or soil digestate analysis and plant tissue analysis did not reveal significant differences in plant uptake between the group one and control columns as the SRS soil contains native ^{133}Cs and ^{238}U.

***Keywords:** Plant uptake, Environmental Transport, Non-human biota*

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PS1 (T1.4-0993)

Effects of Vitamin E on the Radiation Protector and Tumor growth during Radiotherapy

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The purpose of this study is to evaluate the effects of vitamin E (VE) on the immune system and tumor growt during radiotherapy (RT) in mice model. C57BL/6NCrSlc mice were randomly distributed in four grouṗ (control, VE alone, RT alone, and VE+RT). In the VE and VE+RT groups, VE was administered in the diet : 500mg/kg. Radiation was delivered at 2 Gy in a single fraction on the whole body or at 6Gy in three fractior locally in the RT and VE+RT groups. Changes in leucocytes and T lymphocytes were counted and compare between the four groups. The number of leukocytes was increased in the VE group compared with that in tl control group. The magnitude of leukocyte recovery after RT was also increased by VE. This change wa affected largely by alterations in lymphocytes and monocytes rather than that in granulocytes. Both CD4+ an CD8+ T lymphocytes were positively affected by VE. To evaluate the effects on tumor growth, Ehrlic carcinoma cells were injected into the thighs of mice, and tumor volumes and growth inhibition rates wer compared. The tumor growth was inhibited not only by RT but also by VE alone. If RT was delivered wit VE, tumor growth was markedly inhibited. In conclusion, VE could increase the number of leukocyte primarily lymphocytes, even after RT was delivered. VE also inhibited the tumor growth in addition to R Thus, VE may be a useful radioprotective supplement in radiotherapy without inducing tumor growth.

Table 1. Tumor weights and growth inhibition rates following vitamin E administration.

Groups	Tumor weight (g)	Tumor growth inhibition rate (%)
Control	1.28 ± 0.55	-
VE	0.29 ± 0.20	77.1
RT	0.05 ± 0.02	95.8
VE+RT	0.01 ± 0.01	98.8

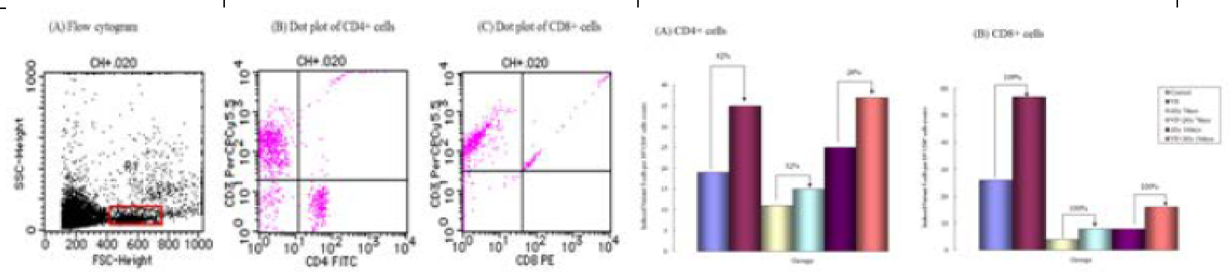

Fig. 1.Changes in CD4+ cells (A) and CD8+ cells (B) after exposure to 2Gy whole body radiation and vitamin E administration. Effects of vitamin E on tumor growth in unirradiated groups (A) and irradiated groups (B) inoculated with Ehrlich carcinoma cells. Statistically significantly different (* P<0.05, ** P<0.01 from the control group.

***Keywords:** Vitamin E, radiotherapy, immunity, lymphocytes, tumor*

ACKNOWLEDGMENTS

The authors would like to thank the Dr. V. Kagiya form Kyoto university.

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PS1 (T1.4-1119)

The European Radioecology Alliance has updated its Strategic Research Agenda and joins the European Radiation Protection Roadmap

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The European Radioecology Alliance (ALLIANCE) was established in 2009 from the conviction that joining forces would enhance the competence of radioecology science in Europe. The main objective of ALLIANCE is to progressively strengthen the coordination and integration of research in the field of radioecology at national, European and international level. Today, the ALLIANCE is 31 members strong coming from 16 countries.

ALLIANCE, in close collaboration with the European Network of Excellence in Radioecology STAR (Strategy for Allied Radioecology) and the European 7FP project COMET (COordination and iMplementation of a pan-European instrumenT for radioecology), has developed for the first time a Strategic Research Agenda (SRA) on Radioecology. A number of different drivers steered the development of the SRA such as credibility concern, new paradigms and scientific advances; potential risks, growing awareness by the public of the importance of the global quality of environmental resources and biodiversity, the need for an integrated approach in order to improve the degree of realism in dose assessments, the need to develop applied research activities in order to solve recommendations and requirements of the new EURATOM BSS that are related to radioecology.

The SRA identifies three challenges: (1) To predict human and wildlife exposure more robustly by quantifying the key processes that most influence radionuclide transfers; (2) To determine ecological consequences under realistic exposure conditions and (3) To improve human and environmental protection by integrating radioecology. It identifies 15 research lines associated to the 3 above challenges, consistent with a strategic vision of what radioecology can achieve in the future via a prioritisation of efforts. The initial full version, released in 2013 after incorporation of more than 100 comments from a wide panel of stakeholders, is available at www.radioecology-exchange.org.

The integration of the European radioecology community through the ALLIANCE in the European Radiation Protection scene is key in the context of the EURATOM Horizon 2020 and Horizon Europe framework programme. ALLIANCE is an active partner, together with the other radiation protection platforms MELODI, NERIS, EURADOS, EURAMED and the newly established SHARE within CONCERT, a European Joint Programming Instrument for radiation protection under which radiation protection research in Europe is managed. One of the tasks within CONCERT has been the update of the platforms SRAs. The updated version considered the advancement in science through CONCERT and affiliated projects, via other European research initiatives and progress within the ALLIANCE topical working groups (Marine radioecology, Human food chain modelling, Naturally Occurring Radioactive Materials (NORM) Radioecology, Transgenerational effects and species radiosensitivity, Atmospheric dispersion and transfer processes). It considered the comments of the ALLIANCE External Scientific Advisory Board (ESAB), research needs in support of the implementation of the EURATOM BSS, ICRP research priorities, the gaps analysis elaborated in 2018 under CONCERT, new challenges presenting and highlights increased interaction with the other radiation protection platforms. Key changes and key remaining scientific challenging (and challenges related to Education and Training, Infrastructure and Stakeholder Involvement) will be articulated during the presentation.

Keywords: *Radioecology, Research Plan, Radiation Protection*

Poster Session T1

PS1 (T1.5-0751)

Ecosystem approach to multispecies experiments in radioecology - literature review

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Multispecies experiments like microcosms and mesocosms are widely used in many fields of research but are uncommon in radioecology. In radioecology, size limitations are important with regard to problems with radioactive waste or shielding, absorption and available space (indoor gamma fields). This literature review (ISI web of science, n=406) assess design and properties of multispecies effect studies (<100 L and >2 species) across research fields to assess their suitability to radioecology. This poster presents main findings and recommendations of the review published in Science of the Total Environment in 2019. Almost all microcosm or mesocosm (cosms) studies assess some ecosystem level parameter. Studies with more taxa assess structural ecosystem parameters more often, while the opposite trend is seen for indirect effects/interactions. Most cosms are custom-made rather than standardised designs. Unmanipulated cosms consist of excised portions of the natural environment with a higher number of mentioned taxa, high ecological complexity and high realism, but have a relatively low replicability. In contrast, standardised cosms with fewer taxa have less ecological complexity but much higher replicability. However, even very small cosms can be ecologically complex (e.g. number of taxa and trophic levels) and stable over time. We encourage cosm radioecology studies, preferably with environmental relevant doses and sufficient detail on dosimetry.

Keywords: *Ecosystem approach, Radioecology, Indirect effects*

ACKNOWLEDGMENTS
This work was supported by the Research Council of Norway through the Centers of Excellence funding Scheme, project number 223268/F50, through the CERAD Centre of Excellence.

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PS1 (T1.6-0739)

Implicit attitudes about Agricultural and Aquatic Products From Fukushima depend on Where Consumers reside

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Key words: Implicit association test (IAT), hesitant purchase, Fukushima agricultural and aquatic products

Japanese consumers are still hesitant to purchase products from Fukushima, although 8 years have passed since the Fukushima nuclear disaster and these products are officially considered safe. In this study, we examined whether Japanese consumers have negative implicit attitudes towards agricultural and aquatic products from the Fukushima region and whether these attitudes are independent of their explicit attitudes. Japanese students completed an implicit association test (IAT) and a questionnaire to assess their implicit and explicit attitudes towards products from Fukushima relative to another region. The results reliably demonstrated that the public has negative implicit attitudes towards Fukushima products, whereas their explicit attitudes are consistently positive. These observations predominantly held for participants living close to Fukushima (Tokyo) as opposed to participants living far away (Hiroshima).

Methods

Twenty Japanese student in Hiroshima and 20 Japanese students in Tokyo participated in the experiment.

First, they completed the typical IAT, which is designed to measure the implicit attitudes that may exist on an unconscious level. For a computer–based IAT, a participant was asked to complete a series of several tasks. In the first task, the participant was asked to sort a *category* of each image depicting an agricultural product labeled with "Fukushima products" or "Saga products" by pressing either the left or right key. In the second task, the participant sorted an *attribute* of the presented word, that is, whether it had good or bad meaning. In the third task, the categories and attributes from the first two tasks are combined. For example, "Fukushima products/Good" might require the left response, while "Saga products/Bad" might require the right response. In subsequent tasks, the pairings were repeated and, then reversed (e.g., "Saga products" might be on the same side as "Good" and "Fukushima products" might be on the same side as "Bad"). If the participants could complete the sorting task more quickly and accurately when "Saga" and "Good" were paired than when "Saga" and "Bad" were paired, it suggests that they prefer Saga products over Fukushima products (Nosek, Greenwald, & Banaji, 2008).

After that, participants explicitly reported how strongly they prefer Fukushima products to Saga products with 7-point Likert scale.

Results

Following Greenwald, Nosek and Banaji (2003), we calculated the mean IAT score for each participant group (Figure 1). The IAT score was lower than zero for participants in Tokyo ($B_{10} = 4.67$), but it was not different from zero for participants in Hiroshima ($B_{10} < 1/3.00$).

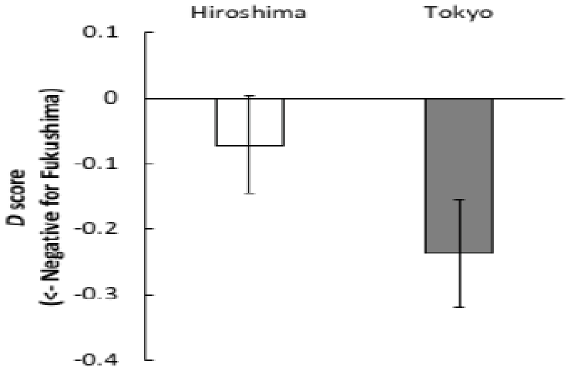

Figure 1. The IAT scores for Hiroshima vs. Tokyo. The lower score means that participants had negative implicit attitudes on "Fukushima" relative to "Saga."

Discussion

The IAT scores demonstrated a reliable negative bias toward Fukushima products, whereas the explicit attitudes tended to be positive. Furthermore, this tendency was pronounced for Japanese living in Tokyo as compared to Japanese in Hiroshima. These results suggest that consumers' implicit attitudes on Fukushima agricultural and aquatic products depend on residential areas, which might potentially underlie the hesitant purchase for Fukushima products.

PS1 (T1.B-0441)

Radiation Safety Compass: Increasing Stakeholder Involvement with Online Tools

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The use of ionizing radiation in medicine, energy production, industry, and research brings benefits to workers, patients and members of the public when used safely. The public and other stakeholders may have incomplete knowledge or may need further information to understand the presence and effects of radiation of natural or artificial origin, and the related radiation protection measures.

As highlighted by the IRPA conference's theme, building transparent and open public communication and empathy with the public might be a key challenge for the radiation protection community. Delivering timely, understandable and technically sound information with the help of new technologies might provide efficient, easy-to-use, flexible, measurable and resourceful ways to reach the public. While face-to-face interactions and traditional communication methods remain important, contemporary communication techniques such as online platforms and social media used also in radiation protection have increased in speed and volume. Technological developments continue to advance the range of options available to those communicating about radiation protection with half of the world population using Internet.[1]

Online tools, identified as an integral part of a communication strategy in IAEA Safety Standards Series No. GSG-6, can increase public trust and confidence in regulatory bodies and radiation protection community in general by keeping the public and other interested parties informed in a timely, transparent and open manner.[2] The IAEA is utilizing and creating numerous online tools supporting radiation professionals in better communication of radiation protection such as Q&A online web pages for patients and health professionals visited 500 000 annually, and monthly radiation protection webinars attended by various stakeholders from 133 countries.

Member States continue to express interest in, and willingness to learn from the relevant IAEA experience. As a result, several member States have approached the IAEA with requests for developing additional online tools, for making the existing IAEA online tools and templates available for their own use, and for providing a guidance on how to produce such tools. Based on these requests, the IAEA has launched a new project Radiation Safety Compass providing directions for radiation professionals in the online communication.

The new online platform called Radiation Safety Compass aims to fil the gap of online communication guidance in radiation protection as a publicly available training resource. It serves as a practical instrument for regulatory and other governmental bodies but also for radiation protection professionals providing information and specific tools to support online communication. The Compass identifies key messages, key considerations when communicating online, but also provides recommended online tools to develop and good practices from the IAEA and from Member States.

The Compass provides answers to the following questions:

- What are the challenges in and tips for communicating specific radiation protection areas online?
- What are the appropriate online communication channels/tools to use?
- How can the regulatory bodies establish these channels/tools?
- What are the best practices?

Keywords: Public communication, Radiation protection, Online communication

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PS1 (T1.B-0466)

Professional Ethics and Effective Communication are Two Sides of One Coin in our Interaction with the General Public

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The recent cultural and societal evolution necessitates for radiological protection professionals to recognize that the general public is no longer satisfied with the scientific reasoning and the technical explanation of the complex decision-making process in radiological protection. Our profession has to earn the trust of our audience. This requires us to exhibit the integrity requisite of ethical professionals to gain acceptance of our recommendations which are scientifically and technically sound but may be difficult to explain. Our work encompasses a broad range; recommendations for protective measures during a radiological emergency may require a different message than the education of the general public about the benefits and hazards of applications of ionizing radiation, e.g., in radiology. However, our basic professional conduct needs to be the same. While ethical professional conduct is essential to achieving trust, effective communication further supports its development and maintenance in our interaction with the general public. We view professional ethics and effective communication as two sides to one coin: the establishment and retention of public trust in our work as radiological protection professionals and our genuine intent to provide for public and individual safety in the use and application of ionizing radiation. This paper develops a concept for and describes the requisite components to an ethical decision-making process by a radiological protection professional, their responsibilities, duties, and virtues, and provides a novel approach to allowing for effective communication of the justification for decisions and their scientific basis. Our approach to communicating with the general public needs to be tailored to the individuals in the audience to open channels for the transfer of information; it needs to motivate and retain the audience for active listening. This implies that we first need to justify to the audience "why" they should consider "our facts," before we can expect them to trust our attempts to provide educational information or to accept our recommendations on "how" to respond to a given situation.

Keywords*: ethics in radiological protection, professional ethics, effective communication in radiological protection, communication with the general public*

PS1 (T1.B-0553)

Society for Radiological Protection (SRP) Public Outreach: Building Radiological Protection understanding through public engagement

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The Society for Radiological Protection (SRP) is the UK's professional and learned society and IRPA affiliate for the field of Radiation Protection. Founded in 1963, and now with ~2500 members, SRP and was incorporated by Royal Charter in 2007 with the mission statement to "promote the science and art of radiation protection and allied fields for the public benefit".

To this end we have an active outreach program. In 2012 SRP started to actively expand its outreach to the future generation of RP professionals. This started when they organized a 'Schools event' as part of the IRPA 13 Congress in Glasgow. In the following years there were a number of smaller events associated with the SRP Annual General Meetings. A review of these events was published in the Journal of Radiological Protection in 2015[1].

This review and forward plan re-enforced SRPs commitment as the chartered Society for RP to expand this engagement and in the UK in 2016 the SRP took the plunge into Public Outreach with their first stand at Big Bang Science Fair in Birmingham's National Exhibition Centre. This is a 4 day event with around 70,000 school children for all over the UK attending and was a significant increase to SRP's commitment to public outreach and engagement. Since then SRP has continued to attend the Big Bang Fair annually and push its public outreach programme further.

At the ongoing public outreach events, SRP talk to children about the basics of the science behind RP, by delivering a range of engaging activities (such as simulating shielding with foam dart guns) and answering questions on the SRP Schools Posters which are displayed and used to run a quiz. Each activity has a supporting lesson plan free to download from the SRP website (https://srp-uk.org/public-and-schools/resources-for-schools), so any teachers or youth group workers can deliver the activities themselves to their young people.

Key to sustaining these activities is funding and resources (including volunteers). Noting these challenges SRP have been looking at efficiencies to reduce these cost burdens, whilst maximizing outreach and developing metrics in order to measure the effectiveness of these events. However a key future work stream is finding alternative funding opportunities and avenues for public engagement. Could Social Media be the key?

The proposed talk will explore SRPs outreach activities, and initiatives to take this forward. We hope this will be of interest to attendees involved in educating and inspiring the next generation of workers and the public in radiation protection.

Keywords: *Outreach, Public Engagement*

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PS1 (T1.B-1058)

Two-Way Effectiveness Evaluation in NST Education and Its Case Study

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Nuclear science and technology (NST) contributes to the health and wellbeing of mankind through the peaceful application in various areas. The sustainable application of NST requires a strategic approach for ensuring the availability of nuclear professionals across all generations. As part of achieving continuity in nuclear workforce across the world and generations, it is important to develop the school teachers' understanding of NST topics and to equip them with creative and exciting methods, tools, and ways in which to deliver such topics to their students. In training course for achieving this purpose, we think "two-way communication" and "feedback" are some of the essential keywords and two-way evaluations during training where lecturers test the participants grasp of the contents and the participants evaluate the presentations are effective. We hosted a two-week workshop (TTWS2019JPN) for secondary school teachers under the technical cooperation program, RAS0079, of the International Atomic Energy Agency (IAEA) and 16 participants from 12 countries were in attendance. We conducted two-way evaluations that were designed not only to gauge the participants' understanding and opinion of the lectures, but also their self-assessment of level of prior knowledge of the content and the impact of the lectures. The test for the participants' grasp of the contents were held twice using same questions without any opportunity to access the questions during the period between first and second tests. The first test was given immediately after each lecture by using a mobile-coupled assessment application called "Plickers™". The second was at the end of the workshop by paper test. The participants' self-assessment was conducted using questionnaires in which they could assess their own level of knowledge on the lecture content prior to the lecture and whether, after the lectures, they felt that they would need further explanations. Fig.1 shows the analysis result for each lecture where several patterns appear. Pattern IV and VIII are over 50% in some lectures. Pattern IV means that participant didn't have knowledge relating to that topic before the lecture and they thought this workshop is good chance to get the knowledge, and they selected the right answer at the both tests. Pattern VIII means that participant had prior knowledge of the topic before the lecture, and they selected the right answer at the both tests. These results show that participants knowledges before the lectures depend on the topic and this workshop was effective for them to get new knowledges.

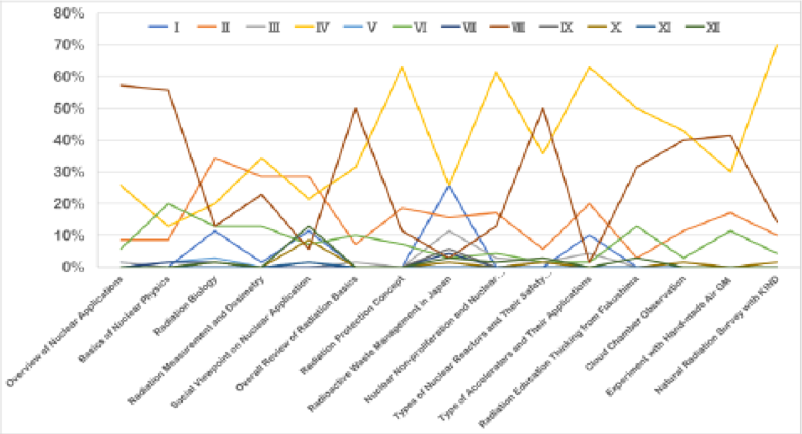

Fig. 1. The analysis result for each lecture of the combined these data in several pattern.

***Keywords:** Education, Nuclear science and technology, Two-way communication, Feedback*

PS1 (T1.D-0853)

Enhancing Radiation Protection Culture through a Gender Perspective

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The concept of culture involves ideas, habits, behaviours, knowledge, experiences, and attitudes, which are developed, shared and accepted by people in a society; and it includes both scientific and social dimensions. The main purpose of this paper is to raise awareness on the relevance of a gender perspective when understanding the development of every culture and, in this particular case, of the construction of a Radiation Protection (RP) culture. In this context, like most nuclear science-related disciplines, RP has historically been a male-dominated field and therefore, policies and strategies have been discussed, designed, and established primarily by a masculinized approach.

In spite of the general agreement on the relevance of diversity, women, while constituting over half of the world population, still remain underrepresented, especially in decision-making processes. In order to consolidate a RP culture, it is essential to integrate a gender perspective that ensures not only an active engagement of women, but also the identification and processing of their differentiated needs and visions, so that they constitute real, key components of this process.

In Radiation Protection, there is concrete evidence that women face obstacles mainly at four levels: socio-cultural, institutional, female subjectivity, and no gender solidarity. Firstly, there is a socio-cultural construction on the role women should play in a society that keeps them away from decision-making positions, and developing professional careers in Science, Technology, Engineering, and Mathematics (STEM) fields. This aspect is related to the female subjectivity, which is also socially constructed, and is part of the collective imaginary and of women's themselves who are raised to be mothers and carry out domestic tasks, according to the established stereotypes. Lack of self-esteem, fears, or insecurities are also elements of the education they receive from kindergarten. As of the institutional level, most regulations, statutes, organizational charts, are elaborated by and for men, so women naturally enter in a more hostile environment, where the lack of gender consciousness prevails. Last but not least, our society has historically promoted enmity among the diversity of women, so that they have not been able to develop strong tools for networking or teamwork among them.

In addition, applying a gender lens to RP is also crucial to cope with one of its most important challenges: public communication and empathy, which is essential for an effective implementation of RP measures. Women also remain having the more negative perception on nuclear, and especially on the effects of ionizing radiation. Thus, in order to bridging the gap of understanding between experts and the society as a whole, action plans need to include dedicated strategies, with its corresponding budget, for developing an innovative narrative and communicating from a gender approach.

In conclusion, it is vital to engage all key players, including high-level authorities, and educate them on the importance of gender mainstreaming in this particular field, considering that the above-mentioned obstacles have a direct negative impact not only on reaching gender parity, but also on the enhancement of a RP culture, and on achieving the required public acceptance.

Keywords: Radiation Protection Culture, Gender Mainstreaming, Public Understanding.

ACKNOWLEDGMENTS
Special thanks to Ms Marina Di Giorgio who have really motivated and supported me to further study this topic by focusing my efforts on the particular characteristics and impacts on the RP field.

PS2 (T2.1-0029)

Calibration Curve for Harshaw Thermoluminescent Dosimeters (TLD) for the Dosimetry Laboratory of the Ministry of Energy and Mines of Guatemala

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To estimate the effective dose for whole-body dosimetry, it is necessary to calibrate the personal dosimetry system. The purpose of calibrating thermoluminescent dosimeters is to ensure that the cards will give the same response to a given radiation exposure. A population of 120 thermoluminescent cards was irradiated at 5.19 mSv with a beam of Cs-137, calibrated in terms of personal dose using a Hopewell G10-1-20 irradiator. The cards were measured in the reader Harshaw 6600 PLUS and the reading is based in terms of electric charge. The cards that presented 2% of variation or less than the average, which we corrected by the homogeneity factor (ECC), were chosen as calibration cards. These cards were irradiated with six different values of doses using an ISO water phantom. After reading the cards, the relation between dose and charge resulted with a calibration factor of value (0.0262 ± 0.0002) mSv/nC. With this value, we can calculate an unknown dose received and we can monitor doses of personnel working with radiation.

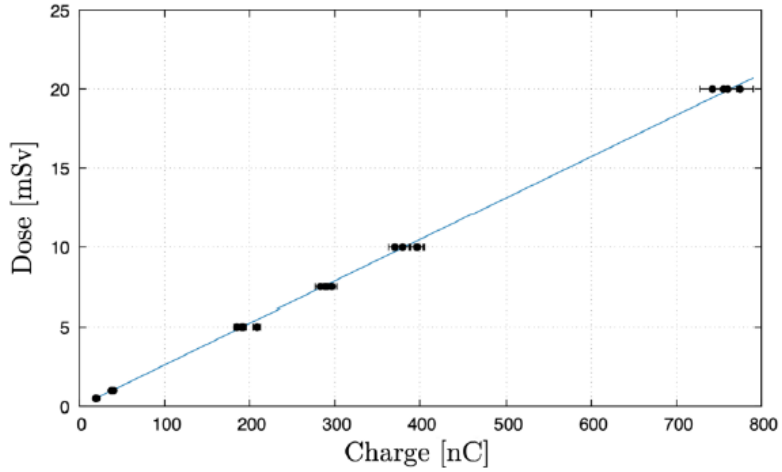

Fig. 1. Dose and Charge.

***Keywords**: Dosimetry, Calibration, Thermoluminescent*

PS2 (T2.1-0035)

Determination of surface skin dose increase for IMRT breast patients using CIVCO breast support thermoplastic system

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For breast cancer patients and IMRT technique, it is used for breast support the CIVCO thermoplastic MTRF19182.4. This support covers all the patient breast volume so the patients keeps in the same position over all the treatment, this system is used for patients with big and pendular breast. The thermoplastic material is fixed to a breast support, it has 2.4mm thickness and it is in contact with the patient skin, this can produce a doses increase at the skin surface.

In the present work it is been measured the dose increase to the skin surface due to the thermoplastic system support, for breast IMRT treatments; gafchromic EBT3 film has been used to evaluate the "build up" region for the thermoplastic material. It was measured for the skin dose 76±1.5% relative to the maximum dose.

Keywords: *Radiotherapy breast, thermoplastic shell, skin dose*

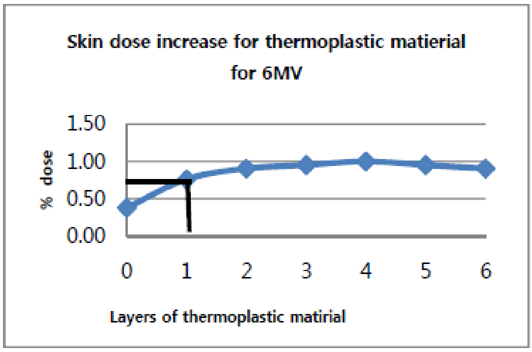

Fig. 1. "build up" for 6MV photons for thermoplastic de 2.4 mm thickness material

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Clínica de radioterapia La Asunción, Guatemala

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PS2 (T2.1-0044)

Method for improving airborne alpha concentration monitoring

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Monitoring of airborne alpha-emitting radionuclides is aimed to assess doses to workers arising from exposure due to intakes of such nuclides and estimate potential exposures for members of the general public. For aerosols, a widespread technique is to pump an air volume through a filter for a given period and then measure the filter at a portable sample counting detector. However, determining the activity collected may be a challenge due to the influence of several factors, one of those is that alpha radiation is extremely limited in its ability to penetrate matter. The self-absorption of airborne alpha-emitting radionuclides is an important phenomenon that occurs in the filter medium.

This paper aims to identify and evaluate the different elements that can affect the direct measurement of alpha-emitting radionuclide's collected in a filter medium and suggest a method for weighing the associated uncertainties. The principal task was to acquire some air filters from a nuclear facility that handles alpha-emitting radionuclides, to measure their 'apparent' activities (i.e. from direct monitoring) and their true activities (from radiochemical analysis) and to derive a ratio of 'apparent to true' activity for different samples.

The results obtained showed that the influence of humidity for a direct measure depends on the workplace area and the origin of the samples. The dust loading on the filter medium, the distance between the paper filter surface and the detector and other uncertainties affect the direct measure.

The correction factor obtained by this paper can only be applied in this particular case. Each facility should make its evaluation, to improve and strengthen the radiological protection of workers in correspondent facilities.

KEYWORDS: airborne alpha concentration, direct monitoring, method, improve

PS2 (T2.1-0061)

Organ Dose Determination in Femoropopliteal Artery Angioplasty and Stenting Procedures Using Monte Carlo

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Obstructions of the femoral and popliteal arteries present the most frequent indications for percutaneous transluminal angioplasty. Femoropopliteal is noted to have proximal femoral occlusions and complete occlusions of the superficial femoral artery as two primary indications. In about 2% of the cases, puncture of the deep femoral artery is performed under fluoroscopic guidance; a contrast agent is then injected to visualize the arteries.

The aim of this study was to estimate organ and effective doses associated with femoropopliteal angiography using Monte Carlo program to simulate the procedure. Data of 21 patients (Age 68.0±13.4 yrs; BMI 26.6±4.9 kgm^{-2}) who underwent femoropopliteal angiography procedures within a 14-month period, were extracted and simulated using PCXMC version 2.0.1.4. Input data for the simulation included patient height and weight at the time of the procedure, image height and width, projection angle, distance source-to-detector, distance source-to-patient, dose area product, X-ray tube potential, filtration and the anode angle of the X-ray tube.

Simulated organ doses from the study (Table 1) indicate that the skeleton received highest dose per patient of 2.3 mGy (0.4 – 11.0 mGy). The single most high individual organ dose of 14.5 mGy recorded in the study was delivered to the uterus. Most of the organ doses were found to be below the 90th percentile (P90) mark of the dose distributions recorded in the study. Organs such as the brain, breast, extra thoracic airway, heart, lungs, oesophagus, oral mucosa, salivary gland, thymus and thyroid recorded infinitesimal dose levels. The estimated wholebody dose per patient of 1.4 mSv was found to be three-fold as much as the effective dose estimated based on ICRP 103 protocol.

Table 1. Simulated organ doses (mGy) from femoropopliteal procedures

		Min	Max	Mean ± δ	P90	Sum
Organ Dose (mGy)	Active bone marrow	0.04	9.14	1.15 ± 2.44	2.21	22.91
	Adrenals	0.00	0.34	0.03 ± 0.08	0.09	0.56
	Colon	0.03	6.59	0.94 ± 1.83	1.79	18.74
	Kidneys	0.00	1.03	0.11 ± 0.28	0.18	2.10
	Liver	0.00	0.17	0.02 ± 0.05	0.05	0.41
	Lymph nodes	0.02	2.42	0.30 ± 0.64	0.50	5.94
	Muscle	0.25	5.95	1.44 ± 1.86	5.44	28.82
	Ovaries	0.01	12.54	1.65 ± 3.36	2.85	33.07
	Pancreas	0.00	0.30	0.03 ± 0.08	0.12	0.64
	Prostate	0.13	8.35	1.72 ± 2.63	6.43	34.34
	Skeleton	0.44	10.97	2.26 ± 2.93	7.71	45.14
	Skin	0.28	6.48	1.47 ± 1.81	4.85	29.37
	Small intestine	0.00	7.85	0.82 ± 2.07	1.04	16.35
	Spleen	0.00	0.37	0.04 ± 0.09	0.09	0.69
	Stomach	0.00	0.53	0.05 ± 0.13	0.07	0.97
	Testicles	0.08	3.03	0.69 ± 0.88	1.68	13.71
	Urinary bladder	0.05	7.17	1.23 ± 2.06	3.94	24.65
	Uterus	0.02	14.50	1.67 ± 3.60	2.70	33.46
Whole body dose (mSv)		0.25	6.19	1.41 ± 1.83	5.26	28.15
Effective Dose ICRP103 (mSv)		0.04	3.25	0.48 ± 0.88	1.09	9.59

Keywords: Femoropopliteal, Angiography, PCXMC

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PS2 (T2.1-0066)

Neutron-gamma mixed field measurements in Hp(10) by means of a TLD600-TLD700 dosimeter pair

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Mixed neutron-gamma field dosimetry still stands a challenge because of the difficulty to experimentally discriminate the dose from each field component. As explained in the ICRU report 26, the use of a suitable pair of dosimeters is needed to discriminate the contributions of gamma photons and neutrons in the mixed field. One of these dosimeters must to be more sensitive to neutrons than the other. The TLD (thermoluminescence dosimeters) ^{7}LiF:Mg,Ti (TLD700) and ^{6}LiF:Mg,Ti (TLD600) are usually chosen for measurements in a thermal neutrons and gamma mixed field. The TLD600 is much more sensitive to thermal neutrons than the TLD700, which interacts very weakly with thermal neutrons. On the other hand, the sensitivity to gamma photons of both types of dosimeters is approximately equal.

In this work, the method applied for neutron-gamma personal routine dosimetry used in the Radiological Protection Division of the Bariloche Atomic Center was analyzed. The theoretical study and its experimental validation of the dosimeter response in terms of the personal equivalent dose Hp(10) were performed. Irradiations were performed with a gamma source, ^{137}Cs, and in a mixed neutron/gamma field of low flux provided by a ^{241}AmBe source. Doses delivered were measured both empirically and verified analytically by Monte Carlo simulations.

The TLD700 reading could be linked to the gamma dose, Hp,γ(10), by applying the gamma calibration factor obtained. In order to associate the TLD600 reading to the neutron dose, Hp,n(10), the contribution of gamma dose to this dosimeter was firstly estimated. The difference between both dosimeters was linked to the neutron dose by applying the neutron calibration factor computed.

The method described in this work would allow to obtain a set of calibration factors in order to know separately the Hp,n(10) and Hp,γ(10); both in the real routine mixed fields as well as in the cases of radiological accidents.

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PS2 (T2.1-0075)

Estimation of the Terrestrial Radioactive Nuclides and Radiological Hazards in Sukari Gold Mine, Egypt

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The Sukari granitoid pluton, the South Eastern desert of Egypt is one of the best examples of gold-bearing granites in the Arabian Nubian Shield (**ANS**), where their rocks are the highly economic kind used for mining gold. We are interested in the current research to estimate the different radiological parameters and assessment hazard indices by using several nuclear techniques. Also, our study improves the abundance of uranium-235 which helps in mining for uranium in the future. On the other hand, we make sure to bind the presence of gold ore with heavy elements. 20 samples were collected from major fractures and thrusts in Sukari gold mine and its surrounding areas (**SGMA**). Passive technique with Solid State Nuclear Detectors (**CR-39**) is used to determine radon concentrations. The active technique was also used for other important radiometric measurements by Sodium iodide scintillation activated with thallium **NaI(Tl)** and Hyper-Pure Germanium **HPGe** gamma-ray spectrometers.

The investigated areas were systematic to many locations as Wadi Fegas, Gabal Rabdi, Sukari North, Gabal Ghadir (SGMA).

The calculated average value of radon concentration is equal to (11.27±0.08 kBqm^{-3}) and that for the annual effective dose is 35.71±3.05 (mSvy^{-1}).The average specific activity of Uranium-238, Radium-226, Thorium-232, and Potassium are: (3381.30), (4415.21), (2752.87) and (2773.46) Bqkg^{-1} respectively.

Also, different radiological parameters such as Radium Equivalent (Ra_{eq}), Hazard Indices ($H_{int.}$–$H_{ext.}$), Gamma index (Iγ) & Alpha index (Iα), Utilization index (I), Excess Life Cancer Risk (ELCR), Exposure Rate (E_R), Dose Rate (D_R), Absorbed Gamma Dose Rate (D), Annual Effective Dose Equivalent(AEDE), Effective Dose Rate (D-organs), Annual Gonadal Dose Equivalent (AGDE). The uranium, thorium, and radium equivalent values were also calculated in Part Per Million (PPM) and the Potassium percent.

The obtained data will be of great help when mining of Uranium in the future and detecting any change in the radioactive background level due to geological processes, and it can be used as a reference information data in Sukari gold mine areas. Strong safety considerations are recommended for protecting the working personnel in surrounding areas from any harmful radiation that would affect them.

PS2 (T2.1-0128)

A workplace trial of eye lens dosimetry at the Atomic Weapons Establishment, UK

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The Ionising Radiations Regulations 2017 (IRR17) specify an annual limit on equivalent dose for the lens of the eye of 20 mSv; this is considerably lower than the previous limit of 150 mSv. A workplace trial of eye lens dosimetry was carried out at the Atomic Weapons Establishment to measure occupational doses and inform decision-making on the requirement for ongoing dose assessment and additional control measures.

32 classified persons from three operational facilities (A, B – manufacturing, and C – component testing) were selected to participate in the trial, based upon fulfillment of one or more criteria indicating a likelihood of receiving a "significant dose" to the lens of the eye under IRR17, interpreted as 1 mSv or greater. The criteria used were as follows: i) effective dose to the whole body of 1 mSv or greater, ii) exposure to weakly-penetrating radiation, iii) exposure geometry with source near to eyes, iv) shielding configuration with less shielding provided to eyes, e.g. glovebox windows. Participants wore approved headband dosemeters provided by the Personal Dosimetry Service of Public Health England whilst carrying out their normal duties within controlled areas for two consecutive quarters, from June-December 2018.

Results from the workplace trial demonstrate that doses to all participants were well below the limit (see Fig. 1). For Facilities A and B, the percentage of participants who received a significant annualised dose was 38% and 27%, respectively, with average annualised doses of 0.90 mSv and 0.70 mSv. The ratio of eye lens to whole body dose was 2.4 for Facility A and 3.0 for Facility B. Ongoing dose assessment has been introduced for certain classified persons in these facilities, with the intention that future results will be used to refine the extent of this requirement. No participants from Facility C received a measurable dose, indicating that ongoing dose assessment is not required. Shielded eyewear has been specified for certain high dose rate tasks to ensure that doses are as low as reasonably practicable.

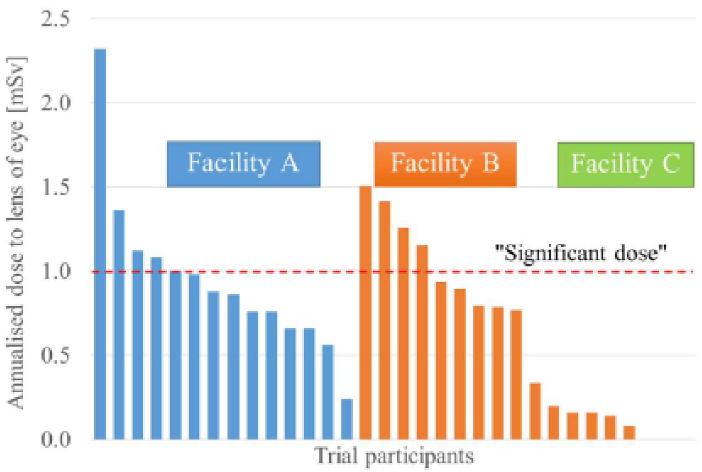

Fig. 1. Annualised dose to lens of eye for trial participants

Keywords: Dosimetry, eye lens dosimetry

PS2 (T2.1-0166)

Effective Dose Estimation and Cancer Risk Assessment of Some Selected Computed Tomography Examinations

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Background

Advances in diagnostic medical imaging in the past few decades using procedures such as computed tomography (CT) have significantly enhanced health care delivery. The effective doses and associated cancer incidence and mortality risks were estimated for adult patients undergoing the five most common types of CT examinations, namely, head, neck, chest, abdomen and pelvis, at Sweden Ghana Medical Centre (SGMC) in Accra, Ghana.

Objectives

The main objective of this research is to determine the effective doses received by selected anatomical region of patients undergoing CT examinations at SGMC and compare them with reference levels (International standards) in order to be able to assess the associated risk.

Design/Method

The two methods employed in the study were patients' data collection and phantom measurements to verify the patients' data. The effective doses were estimated using the dose length product (DLP) from the control console of the CT machine and the anatomic region specific conversion factors. The lifetime attributable risk of cancer incidence and cancer mortality for each patient for a particular examination were both determined from the effective dose, age and sex of each patient with the help of BEIR VII report.

Results

The lifetime attributable risks of cancer incidence and cancer mortality for each patient for a particular examination were both determined from the effective dose, age and sex of each patient using the standard Biological Effects of Ionizing Radiation (BEIR) VII criteria. The effective doses were all within the range of 1 - 10 mSv recommended for CT examinations. The average risk for all the examinations was observed to be very low, i.e. 1 in 10001 to 1 in 10 000.

Conclusion

The study estimated effective doses of anatomical regions and established cancer risks associated with patients undergoing various CT examinations at SGMC. This study was necessary and a step of ensuring patient safety and protection in Diagnostic Radiology.

PS2 (T2.1-0175)

Verification of lithium formate monohydrate in 3D-printed container for electron paramagnetic resonance dosimetry in radiotherapy

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The nondestructive dosimetry achieved with electron paramagnetic resonance (EPR) dosimetry facilitates repetitive recording by the same dosimeter to increase the reliability of data. In precedent studies, solid paraffin was needed as a binder material to make the lithium formate monohydrate (LFM) EPR dosimeter stable and nonfragile; however, its use complicates dosimetry. This study proposes a newly designed pure LFM EPR dosimeter created by inserting LFM into a 3D-printed container. Dosimetric characteristics of the LFM EPR dosimeter and container, such as reproducibility, linearity, energy dependence, and angular dependence, were evaluated and verified through a radiation therapy planning system (RTPS). The LFM EPR dosimeters were irradiated using a clinical linear accelerator. The EPR spectra of the dosimeters were acquired using a Bruker EMX EPR spectrometer. Through this study, it was confirmed that there is no tendency in the EPR response of the container based on irradiation dose or radiation energy. The results show that the LFM EPR dosimeters have a highly sensitive dose response with good linearity. The energy dependence across each photon and electron energy range seems to be negligible. Based on these results, LFM powder in a 3D-printed container is a suitable option for dosimetry of radiotherapy. Furthermore, the LFM EPR dosimeter has considerable potential for in vivo dosimetry and small-field dosimetry via additional experiments, owing to its small effective volume and highly sensitive dose response compared with a conventional dosimeter.

Keyword: Electron paramagnetic resonance, Dosimetry, Lithium formate, Radiotherapy, 3D printer

PS2 (T2.1-0184)

Activity Concentrations of ^{238}U, ^{232}Th and ^{40}K in Medicinal Plants Collected in Southwestern Nigeria and Radiation Dose Due to Their Consumption

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Natural radioactivity levels in forty medicinal plants commonly used therapeutically in a part of Southwestern Nigeria have been measured to determine the annual effective dose due to ingestion of selected natural radionuclides in the plants. Activity concentrations of the selected radionuclides were measured using gamma ray spectrometry technique with high purity germanium (HpGe) detector. Measured activity concentrations of ^{238}U, ^{232}Th and ^{40}K in the plants varied from 0.11±0.02 to 2.38±0.36 Bq kg^{-1}, 0.12±0.01 to 4.19±0.63 Bq kg^{-1} and 4.51±0.68 to 39.84±5.98 Bq kg^{-1} respectively. The annual intake of ^{238}U per person through medicinal plants was calculated (using consumption rate of 0.5 kg y^{-1}) to be 14.56 Bq (equivalent of about 0.15x10^{-2} g). The effective dose equivalent of ^{238}U, ^{232}Th and ^{40}K through the plants ranged from 3x10^{-8} Sv to 39x10^{-8}Sv with a mean of 13.4x10^{-8} Sv and standard deviation of 7.8x10^{-8} Sv.

PS2 (T2.1-0187)

Natural Radioactivity and Radiological risk in granites used for building construction in Ondo State. Nigeria

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Forty samples of granites usually used in building construction in a part of Southwestern Nigeria were assessed for natural radionuclide concentration using a high-purity germanium (HpGe) detector. The activity concentrations of ^{238}U, ^{232}Th and ^{40}K in the granites varied from 7.44±1.12 to 67.83±10.17 Bq kg^{-1}, 17.56±2.63 to 105.33±15.80 Bq kg^{-1} and 309.84±46.48 to 838.87±125.83 Bq kg^{-1} respectively. About 35%, 55% and 80% of the granite samples had activity concentrations of ^{238}U, ^{232}Th and ^{40}K respectively that exceeded the worldwide average values reported by United Nations Scientific Committee on Effects of Atomic Radiation (UNSCEAR) The mean activity concentration of 232Th (54.41 Bq kg-1) and 40K (556.88 Bq kg-1 were higher than their worldwide average of 45 Bq kg-1 and 412 Bq kg-1 reported by UNSCEAR. Radiological hazard indices such as absorbed dose, annual effective dose equivalent (indoor and outdoor), radium equivalent activity, internal and external hazard indices, representative gamma level, internal (α) level, annual gonadal dose equivalent and excess lifetime cancer risk were calculated. More than 25% of the granite samples gave absorbed dose rate, excess lifetime cancer risk, gamma representative index and annual gonadal effective dose values that are higher than their upper limit recommended by international radiological protection organisations, making them unacceptable for use in the building construction industry.

PS2 (T2.1-0228)

Monitoring of Ambient Dose Equivalent at the Boundary of Nuclear Sites to Verify Compliance with Regulations in the Netherlands

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A network of monitoring posts positioned at the boundary of nuclear facilities in the Netherlands measures the gamma ambient dose equivalent. The purpose of this MONET network, operated by RIVM, is to independently verify compliance with the dose limit granted in the operating permit. A combination of proportional counters and Geiger-Müller detectors (fixed as well as portable) are deployed.

The measurements are analyzed in order to distinguish between changes of the gamma dose equivalent rate due to the operations of the nuclear facilities from variations of the natural background. For instance, an increase of the ambient dose equivalent due to the raining out of the radon progeny in air can, to a certain extent, be correlated with local precipitation.

The analysis of the measured dose rates relies on the determination of the variability of the measurements for each of the MONET monitors. This method allows us to estimate, for the measured ambient dose equivalent, the increased radiation dose due to normal operations of the nuclear facilities, together with a confidence limit. The effect of the increase of the volume of radioactive waste and the placing of a wall for shielding purposes are thus clearly identified.

Keywords: *External radiation, radioactive waste, dose limits*

PS2 (T2.1-0285)

Absorbed dose rate evaluation for the Tunisian cobalt-60 irradiation facility by using MCNP5 Monte Carlo code and comparison with GEANT4

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Cobalt-60 (gamma) irradiation facility was mainly designed for sterilization of medical devices and preservation of foodstuff. This tool was put into operation in 1999 at the National Centre of Nuclear Sciences and Technology Sidi-Thabet, Tunisia CNSTN.

A simulation study of the Tunisian gamma irradiation Facility is carried out using MCNP5 Monte Carlo code (X-5 Monte Carlo Team 2003). The purpose of the work was focus to the evaluation of the absorbed dose distribution inside the irradiation cell by modeling a 3-D Monte Carlo model for the gamma irradiator.

In this work, a MCNP model of the Tunisian gamma irradiation facility was developed. Spatial Dose rate distributions were calculated in the air inside the irradiation room.

Simulation results using MCNP5, results using GEANT4 (Gharbi 2005), measured results (Farah K. 2006) were compared, and a good agreement is observed with maximum relative differences less than 5%. This agreement is indicated that the established model is an accurate representation of the Cobalt-60 irradiation facility. It is shown that Monte Carlo simulation improves process understanding, predicts absorbed dose distributions. Our MCNP model can be used for future works such as optimization of irradiation processing and reload of the source.

Key words: Cobalt-60 irradiation facility, MCNP5, Absorbed dose rate, dosimetry.

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PS2 (T2.1-0290)

Effective Radium Contents and Radon Exhalation Rates Corrected for Back Diffusion in Commonly Used Building Materials from South-Western Nigeria

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Building materials which are products from the Earth's crust, have been recognized to be second leading sources of indoor radon after soils and rocks underneath buildings (Amin, 2015) . Measurement of radon exhalation rates with effective radium contents of the materials is a means of assessing the contributions of the materials to the total indoor radon level at home. In this study, the close-can technique, using solid state nuclear track detectors (CR-39) was employed to measure radon concentration, radon exhalation rates and effective radium contents in 173 samples from 26 groups of commonly used building materials from South-Western Nigeria. The measured radon exhalation rates and the effective radium contents were corrected for the effect of back diffusion resulting from varying porosities of the materials in order to generate another set of continuous dependent variables. The effective doses estimated from the set of values corrected for back diffusion were found to be statistically significantly higher than thSose values with no correction using Wilcoxon signed ranked test (Figure 1). The corrected effective doses ranged from 0.005 mSv y^{-1} in Plaster of Paris (POP) to 0.252 mSv y^{-1} in White Marble Tiles (TLW) with an average value of 0.07 mSv y^{-1}. All the values of corrected radium contents in all the samples under investigation were found to fall below the permissible value recommended by ICRP.

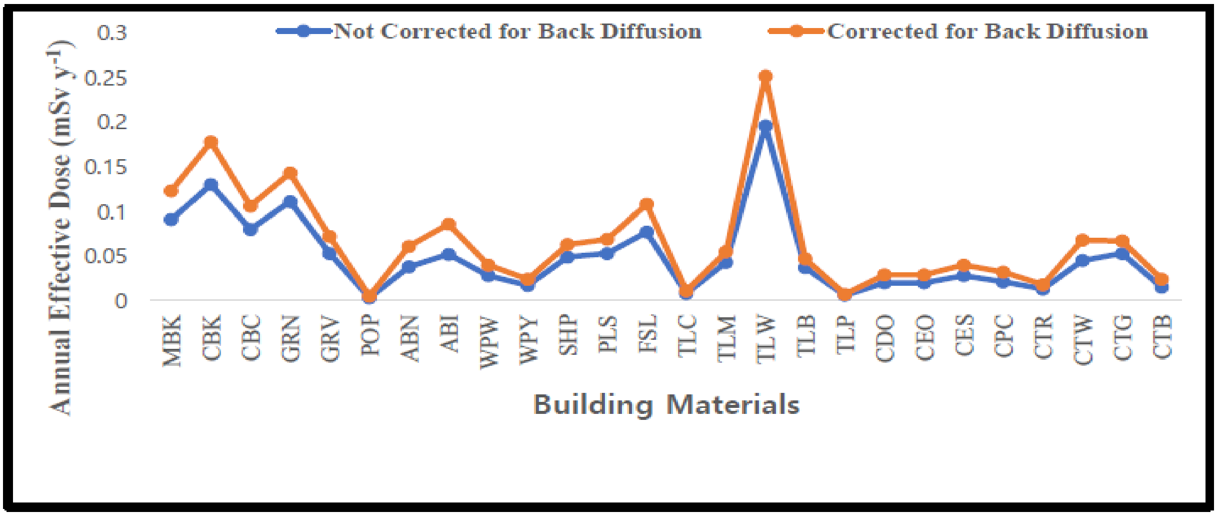

Figure 1: Annual Effective Doses with and without Back Diffusion Correction

Keywords: Building Materials, Exhalation Rates, CR-39.

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PS2 (T2.1-0361)

Dose Assessment According to Various Measurement Conditions for Alanine/ESR Dosimetry to be used in Nuclear Power Plant: Monte Carlo Study

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Evaluation of radiation dose to the insulation cables is essential for safe management of nuclear power plant (NPP) because they are installed in the harshest condition to undergo radiation-induced aging [1]. The International Atomic Energy Agency recommends using alanine dosimeter to qualify ionizing radiation in the NPP. This study aims to evaluate dose to alanine dosimeter with various measurement setups for photons with purpose of deciding standard dose evaluation procedure by using Geant4 Monte Carlo toolkit. Geometry of gamma ray irradiator was modeled with photon energies defined with the data for ^{137}Cs and ^{60}Co supplied by National Institute of Standards and Technology. Cylindrical alanine pellet produced by Magnettech was modeled to assess dose to alanine dosimeter, which has the density of 1.42 g/cm^3. The alanine/ESR dose measurement was simulated as represented in Figure 1 according to different photon beam direction, number of alanine pellets used in each measurement, and distance between the pellets. The number of photon generated in the simulation was decided as 5×10^8. The direction dependence of the alanine dosimeter was evaluated by comparing alanine dose with different beam direction. The dose increment was observed by more than 5% when the photon beam was delivered to the side than the base of the pellet. The alanine dose was observed to increase with increased number of pellets used in a single measurement due to the scattered photon between adjacent pellets. The dose involved with scattered photons was reduced by increasing the distance between pellets. We expect that the results could be used to accomplish improved accuracy of dose measurement using alanine dosimeter and development of standard dose evaluation procedure using alanine dosimeter.

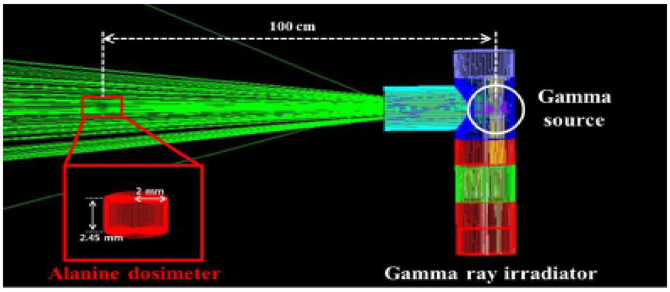

Fig. 1. Geant4 simulation of Gamma ray irradiator and alanine dosimeter

Keywords: Alanine/ESR dosimetry, Safety management, Nuclear Power Plant, Monte Carlo simulation, Genat4

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Poster Session T2

PS2 (T2.1-0367)

Preliminary study to determine a baseline dietary intake of uranium for Necsa workers

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Introduction: The determination of baseline excretion rates due to dietary intake of uranium is necessary in order to evaluate low levels of uranium intakes. The challenge is to differentiate between dietary intake and occupational exposure. Urine monitoring is the most widely used bioassay technique for the monitoring of occupational exposure to uranium. Literature reference several values for typical excretion rates due to dietary intake. These excretion rates ranges from 10 ng U per day to 100 ng U per day. According to ICRP 23, renal excretion of uranium for non-exposed person is in the range between 0.6 mBq/d to 6 mBq/d which corresponds to about 49 ng/d and 490 ng/d. This is a significant large range of values. This varying value complicates evaluation of uranium in urine and a representative dietary baseline study for Necsa workers is needed. The monitoring technique used by Necsa to monitor for uranium intake measure the ^{235}U content (ng ^{235}U) and total uranium in a urine sample. No isotope, specifically ^{235}U, baseline values was found in the literature. It would be preferential to compare the measured ^{235}U against an isotope specific dietary baseline value. The aim of this preliminary study is to determine Necsa specific ^{235}U and Total U baseline values.

Material and Methods: A preliminary study was performed on a non-exposed group which would represent the population mix and eating habits of a Necsa exposed worker. The study was done on historical monitoring data. A section of the monitor workers consists mostly of office workers (e.g. managers, supervisors etc.) who infrequently enters radiological areas and who are monitored for confirmatory reasons. This group were selected as representative of a non-exposed group.

Results: Assuming, dietary intake is natural uranium, the derived ^{235}U dietary intake for the published values (10 ng U to 490 ng U) are 0.072 ng ^{235}U to 3.5 ng ^{235}U which are further derived to a ng/L value i.e. 0.108 ng ^{235}U/L to 5.25 ng ^{235}U/L. The averaged monitored values for the non-exposed group were 4.3 ng ^{235}U/L with some reported values up to 10 ng ^{235}U/L. This is at the upper level of the published values.

Conclusion and discussion: A varying baseline, as found in Publications, complicates evaluation of urine measurements and highlights the importance of a representative baseline. The baseline values determined by this preliminary study are already a better reference value to use than the published values as it is more representative of the Necsa population. However, the non-exposed population are not truly non-exposed seeing that there is a possibility of occupational intakes. A further consideration is do determine an individualised baseline for each individual. However, this would entail monitoring a person during a prolonged period of non-exposure. The data will be further analysed to resolve some uncertainties before a decision will be made whether to use this in the interim until an extended study is performed with non-exposed population.

PS2 (T2.1-0394)

Estimation of Pu/241Am activity in auxiliary lymph nodes and its interferences in direct measurements of lungs/liver activity content

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There is a probability of internal contamination due to Pu/^{241}Am in radiation workers during various processes of nuclear fule cycle in spite of stringent safety measures. In case of injury to hand, radioactivity from the wound gets translocated to auxiliary lymph nodes (LN) and retained there for years.[1, 2] During lung monitoring, Pu/Am activity in these LN can be misinterpreted as lung deposit even if there is no inhalation intake. Whereas, if a person has inhalation intake in addition to wound, then it will be difficult to accurately estiamte lungs and liver deposit as activity present in the auxilliary LN will interfere in lungs and liver measurements. Threfore, a methodology is needed for estimation of auxilliary LN activity and its contributin to the count rate measured for nearby organs, defined as cross talk. In direct monitoring ^{241}Am is used as a tracer and using Pu:^{241}Am ratio, Pu burden is estimated.

In this work, first, actinide monitoring system consisting phoswich detector was calibrated for auxiliary LN geometry and then its cross talk to other organs were estimated. The right and left LN positions were optimized in the Lawrence Livermore National Laboratory (LLNL) phantom in consultation with the experts fromRadiology Department, BARC Hospital, and modifications were made so that ^{241}Am pencil source can be positioned. Measurements were carried out inside totally shielded steel room and calibration factors (C.F.) were evaluated for right and left LN geometries. Cross talk from LN to lungs & liver and vice versa were then obtained by distributing radioactivity in the source organ and no activity was placed in the target organ which was monitored. The estimated C.F. for right and left LN geometry are 2.3 E-02 and 2.1 E-02 cps Bq^{-1}, respectively. If activity is present in the lymph nodes only and measurements are carried out in lung geometry, then about 16-17% of counts recorded in the lymph node geometry contributes to the lungs. If measurement is carried out in liver geometry about 8 % of the right lymph counts contribute to liver whereas contribution from left lymph node toliver is negligible. A methodology was developed to apply cross talk correction in estimation of activity deposited in the target organ. The counts in the organ i, Ci, will have contribution from activity present in all the surrounding organs as

$$C_i = \sum \varepsilon_{i,j} Qj, \qquad (1)$$

Where $\varepsilon_{i,j}$ is the cross talk from jth to ith organ, Q_j is the activity in jth organ. For i=j, ε_{ii} will be C.F. for that organ. Activity Q_j in any organ can be estimated by solving linear equations for various organs. In absence of cross talk correction, activity measured in these organs will be overestimated. The results of this study will be useful

***Keywords:** Auxiliary Lymph Nodes, In-vivo Lung Monitoring, Phoswich detector.*

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PS2 (T2.1-0398)

Guideline to assess the risks of occupational exposure to uranium compounds

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The regional technical cooperation IAEA project RLA9085 was created with the objective of strengthening, in Latin America, capabilities for End Users/Technical Support Organizations on radiation protection and emergency preparedness and response in line with IAEA requirements. In Argentina, the project was separated on interest areas, being occupational radiation protection one of the most relevant.

In the occupational radiation protection area, specific objectives were proposed related to topics that were identified of interest among the Nuclear Regulatory Authority's (ARN, acronym in Spanish) controlled facilities. The specific objectives were oriented to achieve the optimization of protection program and promotion of safety culture in nuclear fuel cycle. In these facilities, many workers are exposed chronically to low levels of uranium in the atmosphere, which results in intakes by inhalation that constitute one of the main contributions to the dose.

The complexity of monitoring and internal dosimetry assessment of workers exposed to uranium is reflected: in the wide variety of compounds to which it may be bound, in the different enrichments and on the importance of taking into account both chemical and radiological risks arising from the exposures to this substance.

The purpose of internal dose assessment in occupational exposure is to provide objective information that contributes to decision-making on follow-up actions in routine and special situations, in order to comply with legal regulations and improve conditions in the workplace.

This paper seeks to present in Latin America a guide to optimize the procedures of monitoring, internal dosimetry and toxicity assessment used for surveillance of occupational intake of uranium compounds that are present in the nuclear fuel cycle.

Keywords: Uranium Compounds, Internal Dosimetry, RLA9085

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Poster Session T2

PS2 (T2.1-0404)

Determination of an environmental background spectrum and data to be used for routine analyses of whole body and lung counts

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Introduction and Background: Radionuclides are found naturally in air, water, soil and in humans. Natural radioactivity is common in the rocks and soil, in our building materials and homes. During the evaluation of incorporated nuclides in workers from occupational exposure one need to differentiate between occupation intakes and natural radiation. A Canberra Whole Body and Lung Counter consisting of 4 BEGE Lung detectors and one scanning NaI Whole Body detector is used to determine the intake of nuclides for Necsa workers. The aim of this study is to determine a representative environmental background spectrum as well as a representative net peak area for each of the identified peaks in the spectrum for different detectors.

Method: Over a period of several months, spectrums were accumulated for different counting times. For each of the peaks in the spectrum the associated natural radiation were identified. The peak centroid (energy) and net peak area fluctuated between spectrums and statistical analyses were done to determine an average peak centroid and an average net peak area. Further statistical analyses were done to determine an upper and lower limit at various confidence levels.

Results: Most of the peaks could be allocated to natural radiation. However, some peaks could not be allocated to a specific nuclide. The coefficient of variance for the net peak area of certain peaks were significantly large. This variation can be attributed to seasonal variances.

Conclusion: A lookup table with representative environmental peaks and associated net peak areas were developed. This table is used during the evaluation of incorporated nuclides for occupationally exposed workers.

PS2 (T2.1-0424)

The design of a radon chamber for the calibration of radon monitors at the Centre for Applied Radiation Science and Technology, Mafikeng, South Africa

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The aim of the study was to design a radon chamber for the calibration of radon monitors at the Centre for Applied Radiation Science and Technology. There are radon monitors in South Africa, however, there are no known calibration facilities in the country. Therefore, there is a need to design radon chambers. The radon chamber was designed with a Perspex material of thickness 6mm and of volume of 0.5 m^3 . Tudor-shaft soil samples whose ^{226}Ra activities were known were used as radon sources. Experimentally; radon concentration, humidity, temperature and pressure were measured with the AlphaGUARDs. The computed radon ingrowth activities were used as a standard for calibrating the experimentally obtained radon activities from radon monitors (AlphaGUARDs). The calibration factors for the experiment were the differences between the radon monitors and the computed radon ingrowth activities at equilibrium determined as 223.97 Bq and 339.83 Bq.

Keywords

Radon monitors, radon chamber, calibration, radon sources, radon ingrowth.

PS2 (T2.1-0439)

MAGNETIC FIELD INFLUENCE ON PERSONAL WHOLE BODY DOSEMETERS

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Keywords:
Personal dosemeter, dosemeter, magnetic field response, occupational exposure

Abstract:
Recent studies of the Panasonic UD-802 personal whole body dosemeters in varying strength magnetic fields (Copty et al., 2019) and an increased demand for understanding of dosemeter response in combined magnetic and ionising radiation fields served as a catalyst for expanding the study to the Mirion dosemeter catalogue.
Testing was performed to determine if exposure to millitesla magnetic fields before, during, and after irradiation resulted in a statistically significant effect on reported results.
The following dosemeters were tested: Genesis Ultra (Harshaw TMCP), APex (Dosimetrics BeOSL), instadose v1.X (DIS), instadose Plus (DIS), instadose 2 (DIS).
Previous type tests have demonstrated that the Mirion dosemeters are not adversely affected by pre-irradiation or post-irradiation exposure to magnetic fields, but no testing has been conducted for concurrent exposure to magnetic fields and ionising radiation.
Future testing should include response validation in stronger magnetic fields.

Copty, A., Rabineg, G., A. Berg 2019. The influence of magnetic fields (0.05 T ≤ B ≤ 7 T) on the response of personal thermoluminescent dosimeters to ionizing radiation. Health Physics Journal. 117(4):345-352.

PS2 (T2.1-0449)

Calculation of organ dose for pediatric patients undergoing computed tomography examinations: a software comparison

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Introduction: The increased number of performed 'Computed Tomography (CT)' examinations raise public concerns regarding associated stochastic risk to patients. Pediatric patients are more susceptible to radiation-induced risks than are adults owing to their rapidly growing tissues and greater post exposure life expectancy. We developed a Dose Archiving and Communication System that gives multiple dose indexes (organ dose, effective dose, and skin-dose mapping) for patients undergoing radiological imaging exams. The aim of this study was to compare the organ dose values given by our software for pediatric patients undergoing CT exams with those of another software named VirtualDose.

Materials and methods: Our software uses Monte Carlo method to calculate organ doses for patients undergoing computed tomography exams. The general calculation principle consists to simulate: (1) the scanner machine with all its technical specifications and associated irradiation cases (Kvp, field collimation, mAs, pitch ...) (2) detailed geometric and compositional information of dozens of well identified organs of computational phantoms that contains the necessary anatomical data. The comparison sample includes the exams of thirty patients for each of the following age groups: new born, 1-2 years, 3-7 years, 8-12 years, and 13-16 years (a total of 150 patients). The comparison protocol is the «Head» protocol. The percentage of the dose difference were calculated for organs receiving 80% of the maximal dose.

Results: The percentage of dose difference between the two software does not exceed 30%. This dose difference may be due to the use of two different generations of hybrid phantoms by the two software.

Conclusion: This study shows that our software provides a reliable dosimetric information for pediatric patients undergoing CT exams.

PS2 (T2.1-0464)

Uncertainties and their consequences in radiation protection

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Uncertainties have important role in several fields of radiation protection. Detailed studies have been performed to identify and characterize uncertainties. Sources of uncertainties and their consequences on the results have been analyzed and will be demonstrated in this paper.

Atmospheric dispersion calculations are used for several purposes in the nuclear industry, not only for environmental impact assessment and safety analysis, but also for decision support systems for nuclear emergency response. For deterministic safety analysis a new applicational method for calculating release criteria with the enhancement of already existing computational formulas has been developed. It was demonstrated that this approach has the additional advantage of reducing uncertainty and providing better transparency and comparability of the safety analysis results. [1]
In the last decade there has been also a growing interest in the probabilistic approach of environmental consequence assessment (Level 3 Probabilistic Safety Assessment) and what benefits and insights can be gained from such analysis to better describe the risks from nuclear facilities, and thus enhance nuclear safety to protect the health of the public.
For decision support systems applicable for emergency preparedness and response ensemble calculations can be used for quantification of the uncertainty originate from meteorological input data. It was demonstrated that application of ensemble meteorological data are suitable to characterize the impact of the meteorological uncertainties on the atmospheric dispersion and dose calculation. [2]
In case of internal dosimetry a new method was implemented in IDEAS Guidelines for quantification of measurement uncertainties by applying a scattering factor. [3] It has been also demonstrated that improving the accuracy and precision of the measurement alone is not sufficient to improve the accuracy of internal dose estimation if the circumstances of the intake are not well known or not available. Uncertainties of the assumptions used in the dose assessment (for example the date and route of intake, the physical and chemical form) can be more influential than the errors of the measured data. It has been stated, that improvement of the accuracy of the analysis would not be an effective way to realize very low uncertainty if the dose consequence is negligible. [4]

Experiences and general lessons learnt from these research areas will be described in this paper. Examples will be presented and consequences will be discussed.

Keywords: personal and environmental dosimetry, decision support systems, safety analysis

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PS2 (T2.1-0485)

External Occupational Dosimetry at the Fuel Elements Manufacturing Plant for Research Reactors

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Fuel Elements Manufacturing Plant for Research Reactors - ECRI, is a nuclear fuel cycle facility of the CNEA - ARGENTINA, where it works with fresh uranium compounds, for the manufacture of fuel elements for research reactors and targets of irradiation for the production of radioisotopes for medical use, from uranium with enrichment less than 20% of $^{235}_{92}U$. ECRI processes approximately around (75 ± 15) Kg of U per year and has the particularity of having 2 internal deposits, which generates accumulation points of radioactive material within it.

The compounds and metal alloys used in the different stages of production are: metallic uranium, $UMo_{(7\%_{(Mo)})}, U_3Si_2, U_3O_8$ and UAl, which they contain mixtures of the isotopes $^{234}_{92}U$, $^{235}_{92}U$, $^{238}_{92}U$, in different proportions according to the enrichment, which can be natural, depleted or enriched at 20%. Additionally, there may or may not be minimal traces of the isotopes $^{236}_{92}U$ y $^{232}_{92}U$, depending on the origin and purification technique of the metallic uranium. All these isotopes are alpha emitters, with low energy gamma emissions and with very low intensity. However, in the decay chains of each of these isotopes, appear daughter products of uranium decay gamma emitters that could significantly increase the occupational exposure of workers due to external radiation.

Currently, ECRI's monitoring plan consists of performing area and air monitoring; and the dosimetry of the staff consists of periodic urine samples, for monitoring internal contamination routinely. As radioprotection measures, the installation has a closed circuit ventilation system to prevent internal contamination. The radioactive material is worked in a glove boxes during all stages of the manufacturing process where the probability of material dispersion is high. In the stages where the use of glove boxes is not possible, workers wear personal protective elements. All these measures are tools to prevent the incorporation of radioactive material. Based on the fact that the gamma radiation field due to uranium is not relevant, unless it is a storage area for large quantities of material, until now, in the ECRI Plant monitoring plan, the need to perform external dosimetry of personnel on a routine basis is not contemplated.

In the specialized bibliography [2], we have not found the quantitative definition of "large quantities". Taking into account the annual volume of uranium processed in the installation, the accumulation points due to the deposits that it possesses and the characteristics of the material, mentioned above, the need arose to make measurements of external dosimetry of the workers to corroborate in empirically, if it was appropriate or not, incorporate into the ECRI's routine monitoring plan, external dosimetry. To achieve this objective, we selected 9 points in the installation, which correspond to jobs where there is a greater probability of accumulation of occupational dose by external radiation and we make measurements with personal dosimeters (TLD) to know the equivalent dose Hp (10) of the workers, assuming the worst possible scenario: maximum number of hours worked per month and the closest possible location to the source.

In this paper, we will discuss the results obtained and the conclusions we have reached.

Keywords: *External Dosimetry, Radioprotection,*

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PS2 (T2.1-0506)

A comprehensive analysis of eye lens radiation doses during external beam radiotherapy of Head and Neck cancer patients

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Aim: In vivo dosimetry is a recommended procedure in radiotherapy. The present study aims to evaluate and compare eye lens radiation doses in head and neck cancer patients treated with external beam radiotherapy (EBRT) among various RT treatment delivery techniques.

Material and Methods: The present study recruited a total of sixty patients age (range, 35 - 70 years; mean, 55 years) with head and neck cancer. The patients treated with conventional 2-Dimensional Radiotherapy (2DRT) using Telecobalt unit Bhabhatron-II TAW (Panacea Medical Technologies Pvt. Ltd., Bengaluru, India), 3-Dimensional Conformal Radiotherapy (3DCRT) using Oncor Expression (Siemens AG Healthcare, Erlangen, Germany) and Volumetric Modulated Arc Therapy (VMAT)/ Rapid Arc™ using dual arc (1 isocenter, 2 full arc) on Trilogy (Varian Medical Systems, Palo Alto, US) in separate treatment arms. All the patients were planned and treated with conventional fractionation regimes, with a dose delivery of 2 Gy/fraction. The eye lens doses were assessed by placing the dosimeter as close as possible to the eye at a predetermined marked place in contact with orfit. The optically stimulated luminescence (OSL) dosimeters were Al_2O_3:C nanoDots™ (Landauer Inc., USA) with dimension of 10 X 10 X 2 mm. The measurements for each patient have been performed for 3 consecutive RT treatment fractions and average reading was considered as eye lens dose per fraction in order to reduce uncertainty in measurement. All the measurements have been obtained during phase 1 treatment of patients in order to take into account worst case scenario for this kind of planned exposure situations which involve patient treatment with RT. A set of information were recorded from each of the participants.

Results: The average eye lens dose during Telecobalt conventional 2DRT treatment was measured 9.05 cGy per fraction for a mean dose delivery of 200 cGy/ #, i.e. 4.50% of the tumor dose. The average eye lens dose during Linac 3DRT was measured 3.84 cGy per fraction for a mean dose delivery of 200 cGy/ #, i.e. 1.92% of the tumor dose. The average eye lens dose during Rapid Arc™ was measured 1.26 cGy per fraction for a mean dose delivery of 200 cGy/ #, i.e. 0.63% of the tumor dose. The average cumulative dose to eye lens was estimated 316 cGy, 134 cGy and 42 cGy during 2DCRT, 3DCRT and Rapid Arc™ treatment delivery of head and neck cancers. Certainly, the actual eye lens radiation doses were slightly below the average cumulative dose. EBRT of Nasopharynx and maxilla carcinomas treatment found to contribute significant dose to eye lens. The highest radiation dose to eye lens was observed in Telecobalt conventional 2DRT treatment. The possible cause of increased radiation dose is due to large collimator opening of field without conformity of radiation beam using beam modifying device i.e. Multi-Leaf Collimator (MLC) as compared to 3DCRT and Rapid Arc™. The eye lens radiation doses were observed less in 3DCRT and Rapid Arc™ treatment deliver technique due to high conformity of radiation field using MLC. The average Monitor Units (MU) were found less in 3DCRT treatment as compared to Rapid Arc™. However, Varian Rapid Arc™ treatment were found to contribute lowest eye lens radiation dose as compared to Siemens 3DCRT treatment. The probable reason is use of tertiary MLC by Varian which provided slightly more radiation protection to eye lens during treatment as compared to Siemens secondary MLC in machines.

Conclusion: Treatment planning of patient, immobilization devices, beam shaping devices, treatment delivery modalities plays a vital role in reduction and magnitude of eye lens dose.
Keywords: *Eye lens dose, External beam radiotherapy (EBRT), Optically stimulated luminescence (OSL) dosimetry*

PS2 (T2.1-0558)

Gender-based Radiography and Radiation Protection to Promote Girl Child Education in a Crisis and Humanitarian Setting in Northeast Nigeria: Chest Radiography Doses

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Background: Northern Nigeria accounts for the highest number of out of school children. Unfortunately, gender gaps in education still exist. The national literacy rate for females is only 48 % compared to 73 % for males. This is further enhanced by the violent conflict and displacement in northeast Nigeria. Children in crisis and humanitarian settings are powerless politically; they are defenceless with no political standing of their own especially the girl child. Hence, they must rely on adults to protect them. Often this protection is only viewed from the social dimension of gender-based violence among others without recourse adequate radiation protection of the girl child undergoing radiography screening involving the use of ionizing radiation as part of school enrollment programme.

Objective: To determine radiation doses received by school-aged female children from chest radiography for school enrollment. To correlate doses with anthropometric and technical parameters to establish age-specific Diagnostic Reference Levels.

Materials and Method: A cross-sectional study was conducted in a major Tertiary Hospital in Northeast Nigeria. Fifty (50) school-aged female children, < 18 years of age were recruited for this study. Demographic data such as age and gender; anthropometric data such as weight, height, Body Mass Index, and anteroposterior chest thickness and technical exposure parameters such as tube potential (kVp), and tube load (mAs) were obtained. Calibrated thermoluminescent dosimeter chips were placed at the centring point of the patient's chest to measure entrance skin dose generated by the tube kVp and mAs. Mean, median and standard deviation were used to summarise the data while percentiles were used to establish Diagnostic Reference Levels. Pearson's correlation was used to determine the relationship between doses received with anthropometric and technical parameters. Diagnostic Reference Level was set at the 75th percentile of the median patient dose distribution for the various age groups studied. Statistical significance was set at $p < 0.05$.

Results; The results of the established Diagnostic Reference Levels for the various age groups in this study were; 4Years - <10Year, 1.56 mGy; 10Years - <14Years, 1.23 mGy; 14Years - <18Years, 2.12 mGy. Pearson correlation shows statistically significant ($p = 0.044$) moderate positive relationship ($r = 0.353$) between weight with entrance skin dose among the 10 Years - <14 Years age group. A significant ($p = 0.035$) strong negative relationship ($r = - 0.703$) was also observed between entrance skin dose with kVp among the 14 Years - < 18Years age group

Conclusion; Age-specific Diagnostic Reference Level for plain chest radiography marked by variations in measured entrance skin doses was established for school-aged female children. A statistically significant strong positive relationship between kVp and entrance skin dose was observed in this study. This is the first gender-based radiation dose survey established in this locality.

Keywords; entrance skin dose, radiation dose, chest radiography, diagnostic reference level, gender-based

PS2 (T2.1-0578)

Radiometric and elemental assessment of groundwater from Ifako-Ijaiye area in Lagos Southwestern Nigeria

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Groundwater from Ifako-Ijaye was examined for the radioactivity and elemental concentrations. Twenty (20) boreholes (BH) and fifteen (15) hand-dug wells (HD) raw water samples were collected from Ifako-Ijaiye area in Lagos. Radioactivity levels were analyzed using a NaI (TI) detector couple, and elemental concentrations using BUCK200 AAS (model: AOAC, 975.23). The effective dose rates (EDR) and excess lifetime cancer risks (ELCR) were calculated from activity concentration of ^{40}K, ^{238}U and ^{232}Th. The activity concentrations and heavy metals concentrations were statistically correlated. The mean activity concentrations of ^{40}K, ^{238}U and ^{232}Th in the BH water were 8.77±8.40 Bq/L, 1.03±0.19 Bq/L and 0.93± 0.15Bq/L respectively; and 10.12±0.07Bq/L, 0.94±0.34Bq/L and 0.89±0.10Bq/L respectively in the HD well water. The mean annual effective dose (D_E) in BH water were 0.230±0.027mSv/y (adult), 0.332 ±0.031mSv/y (crèche), and 0.570±0.069 mSv/y (infants); and in HD well water, the D_E were 0.267±0.061mSv/y (adult), 0.340±0.090mSv/y (crèche) and 0.570±0.157mSv/y (infants). The mean ELCR were (0.52±0.06) x 10^{-3} (adult), (0.76±0.07) x10^{-3} (crèche) and (1.30±0.16) x10^{-3} (infant) for BH water, and (0.52±0.05) x 10^{-3} (adult), (0.78 ±0.07) x 10^{-3} (crèche) and (1.30± 0.12) x 10^{-3} (infant) for HD water. The heavy metal concentrations in BH water were higher than the World Health Organization (WHO) recommended permissible limits. The ELCR in BH and HD water is higher than the USEPA recommended value of 1.0x10^{-4}. It is therefore, recommended that drinking the borehole and well water from the study area should be discouraged.

Keywords: Radiometric assessment, cancer risk, elemental concentrations, borehole water, well water, Lagos Nigeria

PS2 (T2.1-0602)

Clinical Indication-Based Diagnostic Reference Levels for Pediatric Head Computed Tomography Examinations in Kano Metropolis

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Introduction: Pediatric patients are recognized to be at higher risk of developing radiation-induced cancer than adults because of rapidly growing organs and tissues in children are inherently more vulnerable to cellular damage than those of adults. Internal Commission on Radiological Protection (ICRP) has stated that DRLs specific to clinical indications (clinical protocols) are desirable.

Aim: The study aimed to determine the clinical indication based DRLs for pediatric head computed tomography examinations within Kano metropolis.

Materials and Method: Prospective cross-sectional study was conducted in four hospitals within Kano metropolis. A total of 114 patients who undergo Head CT examinations were investigated. Computed tomography dose index (CTDIvol), dose length product (DLP) and other scan parameters were recorded. Pearson Chi-square was use to compare the relationship between the DRLs of different clinical indications and age groups. Statistical significance was set at P>0.05.

Result: The CTDIvol and DLP values observed for the indications were: 27.5 mGy, 1411.4 mGy cm for hydrocephalus, 28.1 mGy, 1035.3 mGy cm for inner air, 39.6 mGy, 1362.6 mGy cm for routine head and 46.2 mGy, 1663.4 mGy cm for paranasal sinuses. Based on age groups, the CTDIvol and DLP were 30.1 mGy, 1004.3 mGy cm for <5years, 32.6 mGy, 1223.9 mGy cm for 5-10years and 37.6 mGy, 1707.6 mGy cm for 11-15years respectively.

Conclusion: Comparison of the third quartile dose values with other studies shows that there is lack optimization in the present study. The main contributor to high dose was the use of different techniques and the used of protocols for adults in some cases by the operators.

Keywords: *Dose reference levels, Computed tomography dose index, Dose length product.*

PS2 (T2.1-0611)

Determination of radiation-induced translocation frequencies using the three-color FISH method

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For retrospective dose assessment or its indication in case of prolonged / chronic exposure, it is recommended to use FISH-staining of chromosomes, which allows detecting reciprocal translocations [1]. Currently, cytogenetic retrospective dose assessment is carried out mainly using the one-color version of the FISH technique. In this case, a single fluorochrome dye binds to DNA probes complementary to the DNA of selected pairs of chromosomes. In general, most researchers proceed from the hypothesis that the probability of each given chromosome being involved in perestroika depends only on the amount of DNA it contains. However, this provision should be checked more thoroughly. It is also interesting to know how registration of exchanges not only between the three selected FISH-stained and counter-stained chromosomes, but also between these FISH-stained chromosomes themselves, can affect the sensitivity of the FISH method. For this, it is necessary to use, for example, a three-color version of this technique.

The material for this primary cytogenetic study was blood obtained from a cubital vein of one healthy male donor (age: 41 years). Radiation exposure was performed in vitro at room temperature with ^{60}Co gamma rays at doses of 0.10; 0.15; 0.25; 0.35; 0.50; 0.75; 1.00; 1.50; 2.00 and 3.00 Gy (dose rate was 0.5 Gy / min). Also, one sample remained unirradiated to record the control level of chromosome aberrations. Ready-made sets of DNA probes for pairs of whole chromosomes Nos. 1, 4, 12 and Nos. 2, 3, 8 (counter-dye – DAPI) by MetaSystems were used for three-color FISH-painting. It should be noted that both of these sets are close to each other in the total relative DNA content.

The following results were obtained.
1. When using any of the two sets of DNA probes, the frequencies of FISH-recorded translocations in all (stable and unstable) cells did not differ significantly from the same value only in stable cells: $p = 0.724$ and 0.131 for 1, 4 and 12 and 2, 3 and 8 pairs of chromosomes, respectively.
2. When combining all the data regardless of the selected set of DNA samples in the range of all doses of radiation exposure from 0.1 to 3.0 Gy, the translocation frequencies in all and stable cells also did not significantly differ from each other with $p = 0.628$. However, with a narrowing of the dose range (0.75-3.0 Gy), the significance level decreased to $p = 0.114$.
3. When comparing translocation frequencies detected using different sets of DNA probes for 1, 4 and 12 and for 2, 3 and 8 pairs of chromosomes, statistically significant differences were also absent in all and stable cells: $p = 0.343$ and 0.114, respectively.
It should be noted that the number of translocations between FISH-stained chromosomes was generally not very large.

Conclusion. In whole both sets of DNA probes gave similar results in terms of radiation-induced translocation frequencies. But in order to obtain statistically significant results and construct dose-response curves for a retrospective dose assessment using a three-color FISH method, it is necessary to continue this study using the blood of other healthy donors.

Keywords: *gamma-irradiation, translocation, three-color FISH method*

PS2 (T2.1-0634)

Profiling the Natural and Anthropogenic Radioactivity Concentration in Selected Canned Food Products Consumed in Nigeria

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An investigation of the naturally and artificially occurring radioactivity concentration in canned food products consumed in Nigeria was carried out using a High purity Germanium detector (HPGe). Three natural long live radioactive materials of ^{238}U, ^{232}Th and ^{40}K were found while ^{210}Pb and ^{60}Co artificial radioactivity were detected in the measured in 22 brands of canned foods. The measured naturally occurring activity concentrations in the canned foods ranged from 27.02±8.07 Bq Kg^{-1} to 90.14±26.31 Bq Kg^{-1} with a mean value of 53.48 ±15.70 Bq Kg^{-1} for ^{40}K, for ^{232}Th, the values ranged from 3.68 ± .08 Bq Kg^{-1} to 25.48 ± 7.54 Bq Kg^{-1} with a mean activity concentration of 13.72 ± 7.14 Bq Kg^{-1}, while for ^{238}U the values ranged from below detectable limit to 27.92 ± 7.86 Bq Kg^{-1} with a mean value of 17.36 ± 5.43 Bq Kg^{-1}. On the artificial radioactivity, the activity concentration for ^{210}Pb it is 1.89 ± 0.14 Bq Kg^{-1} and 0.40 ± 0.19 Bq Kg^{-1} for ^{60}Co. The computed mean radium equivalent is 37.94 ± 13.00 Bq Kg^{-1}. The five radiological risk parameters examined have mean values that are well below their global permissible limits as reported by UNSCEAR. The average Effective Dose due to annual intake of ^{238}U(^{226}Ra), ^{232}Th and ^{40}K from ingestion of examined canned foods were estimated to be 278.1 μSvy^{-1} for infants , 310.0 µSvy^{-1}for children and 241.3 μSvy^{-1} for adults. A comparison of the annual effective values for the three age groups with their reference limits as reported by UNSCEAR and ICRP were found to be below those values.

Keywords: Radioactivity, anthropogenic, canned food, Nigeria.

PS2 (T2.1-0642)

Commissioning dosimetry and dose mapping of the Tunisian Gamma Irradiation Facility and comparison with MCNP Monte Carlo simulation

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Cobalt-60 irradiation facility has been put into operation at the National Centre of Nuclear Sciences and Technology at Sidi-Thabet in Tunisia. Technical specifications were checked by dosimetry commissioning experiments and compared to Monte Carlo simulation data using MCNP. Installation qualification has been carried out to measure absorbed dose distribution in the irradiation cell and products. Two dosimeter systems were employed for measurements: Red and Amber Perspex and Cellulose Triacetate (CTA). The regions of minimum and maximum absorbed dose within a homogeneous dummy products and the dose uniformity ratio were determined. The products were loaded in carrier by sawdust with a bulk density of 0.114 g/cm3, potato with a bulk density 0.58 g/cm3 and syringe with a bulk density 0.123 g/cm3. The isodose curves and the three-dimensional views were built using the kriging method. An acceptable agreement in most sets between Monte Carlo simulation results and the experimental values.

Keywords: Dosimetry commissioning; Monte Carlo simulation; MCNP6.1; Isodose curves; Dose uniformity process, Gamma-irradiation facility

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PS2 (T2.1-0677)

Direct monitoring of $H_P(3)$ neutron exposure of workers' eye lenses in NPPs

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Aim: The article deals with the design and application of the eye lens exposure monitor placed on person's head when working in selected fields of neutrons and gamma rays in nuclear power plant (NPP).

Methodology: The albedo dosimeter was chosen as an $H_P(3)$ personal lens dose equivalent monitor due to its sensitivity, the appropriate size in terms of its location close to the eyes and its directional dependence of the equivalent dose similar to the lens of workers exposed in the NPPs. The properties of the proposed monitor were determined by Monte Carlo (MC) code MCNP6 by simulating the voxel phantom of the Zubal head into which the eyes, described in detail by analytical relationships, were placed. Calibration of the albedo monitor in $H_P(3)$ was performed in the field of neutrons from Pu-Be and Cf sources using the water cylindrical phantom of the head placed on a polyethylene MIRD torso phantom on which the NBG albedo dosimeters monitoring $H_P(10)$ were placed. The properties of the proposed $H_P(3)$ albedo monitor were experimentally verified in selected locations of the NPP where the activities of the personnel were performed during the operation of the nuclear reactor. The neutron and gamma ray spectra at investigated workplaces of the NPP were determined using a moderation spectrometer and MC simulations.

Results: At both investigated workplaces of the NPP (on HCCs board as well as on the reactor hall with C30 fuel cell containers) similar energy spectra of neutrons significantly scattered from heavy steel technological equipment were measured. This spectrum was used to determine the correction factors which take into account the difference between the measured and calibration neutron spectra of monitors of spatial and personal dose equivalents at the monitored sites of work. Measurements at about 20 working positions on HCCs board and at reactor hall with spent-fuel-cell container (cooled for nearly 5 years) showed that the average ratio of dose neutron equivalents $H_P(3)$ of irradiation of the eye lens and $H_P(10)$ on the worker's chest is close to value 1.

Conclusion: A new measuring instrument for direct monitoring of $H_P(3)$ neutron personal radiation dose equivalents in eye lenses of NPP workers was proposed.

Keywords: Eye lens dose monitoring, Neutrons and gamma rays, Nuclear power plant

Poster Session T2

PS2 (T2.1-0679)

MEASUREMNT OF ENVIRONMENTAL RADIATION LEVELS OF METALLIC SCRAP SITES IN WARRI DELTA STATE NIGERIA

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The objective of this study is to measure the Environmental Radiation levels of metallic scrap sites in the city of Warri and the environs. Six major metallic dump sites were chosen for this work, the monitoring was done by using a Geiger Miller type Ionizing radiation monitor Digilet 100 Radiation monitor. Prior to the study the monitor was calibrated with a Cs-137 Source. The result showed that the background radiation outside the immediate vicinity of the scrap site ranged from 0.10µSv/hr to 0.015µSv/hr.Radiation levels from the used pipe for drilling purposes ranged from 0.043µSv/hr to 0.062µSv/hr. The International Commission of Radiation Protection standards radiation (ICRP) level in habitable environment is 0.114µSv/hr. It was observed that environmental radiation levels measured from the metallic scrap sites were lower than ICRP standard. This showed that the metallic scrap sites in warri and environs do not constitute radiological hazards to human and the environment.

Key words: Environment metallic scrap site, Geiger miller Monitor, Warri

PS2 (T2.1-0694)

Commissioning and operating a new Personal Radiation Monitoring Device (PRMD) service in South Africa: A one-year review of data

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Introduction: In South Africa (SA), it is required that, all radiation workers must be monitored using a PRMD supplied by an approved provider. PRMD technologies have evolved over time from earlier film and thermoluminescent based dosimeters to the current standard being Optically Stimulated Luminescence (OSL) dosimeter technology. The OSL dosimeter technology uses an aluminium oxide crystal structure doped with carbon (Al_2O_3:C) that offers many advantages like high sensitivity, non-destructive readouts and improved environmental stability. Unfortunately, -SA doesn't have an approved PRMD provider that uses OSL technology. The aim of this project is to commission and operate a new PRMD service in South Africa utilising OSL dosimeter technology and to evaluate the data generated.

Materials and Methods: Commissioning a new PRMD service in SA requires: South African Healthcare Products Regulatory Authority (SAPHRA) approval; South African National Accreditation Services (SANAS) ISO/IEC 17025:2017 accreditation and; registration with the National Nuclear Regulator (NNR) National Dose Registry (NDR). The new PRMD service, named Dosimeter Services (Pty) Ltd, has been commissioned using OSL dosimetry technology, readers and Individual Monitoring Lab Software (IMLS) supplied by Landauer ®.

Results: The new PRMD service was successfully licensed and obtained all the required accreditation to operate commercially using OSL technology in SA. The service currently monitors 2102 radiation workers primarily in the private healthcare sector. The personal dosimetry records for the first year of operation indicate that the distribution of radiation workers are 54.7%, 17.5%, 17.2 %, 7.2 % and 3.4% in respectively general theaters, interventional theaters, emergency departments, general radiology and radiotherapy. The highest personal doses were measured in the interventional theatre environments.

Conclusion: The improved sensitivity and accuracy that OSL offers was found to be valuable for Radiation Protection Officers (RPO) and the medical physicists when evaluating if radiation worker doses are within safe levels and when to investigate any irregularities.

Keywords: *Optically Stimulated Luminance (OSL), Personal Radiation Monitoring Device (PRMD), Radiation worker, Dosimeter, Personal Dosimetry*

PS2 (T2.1-0712)

Research on High Sensitive Detection Techonolgy of Trace Nuclides

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Terrorism is one of the major threats to public security in the world and also is a serious challenge to world peace and development.In order to effectively prevent the explosion attack of terrorist, all countries in the world attach great importance to the safety control of explosives, especially the flow control and safety inspection of explosives. Tracer technique is an important solution to solve the problem ,western developed countries have proposed dozens of tracer solutions and invested heavily in research and development,there are many methods for the traceability and security detection technology of explosives , but most of them are in the stage of research.Therefore, researchers are trying to find an effective traceability technique to achieve control and traceability of explosive items.

In order to effectively solve the problem of pre-explosion detection of explosives, a trace amount of radionuclide is added to the tracer marker ,targeted to the rapid detection demand of radioactive trace marker in explosives,a high-sensitivity detection device is develped for the detection of γ ray emitting from explosives and is used for rapid and accurate detection of uncontrolled explosives before explosion. As a part of the intelligent control platform for uncontrolled dangerous goods, the device is applied in the field of dangerous goods control, realizes the rapid detection of uncontrolled explosives, and plays a role in preventing and deterring terrorists in social security.

The high-sensitivity detection device of trace nuclides is mainly composed of detector unit, electronic unit, data acquisition and processing unit, object transfer unit and alarm unit. The block diagram of device design is shown in Figure 1. The device adopts a channel type structure, four large-volume plastic scintillators are enclosed into a ring detection channel, and each plastic scintillator is combined with a photomultiplier to form a detection unit of device. Due to the trace radionuclides in the tracer marks, the measurement of trace radioactive materials is not only affected by the performance of the system instrument itself, but also by the surrounding environment of the workplace, Compton scattering and cosmic rays. In order to reduce the influence of the natural radiation background on the detection sensitivity of the device, a special shielding structure is designed around the ring structure detector to shield the external interference. Experiment shows that the shielding structure can effectively shield the natural radiation and the detection efficiency of device is imporved.

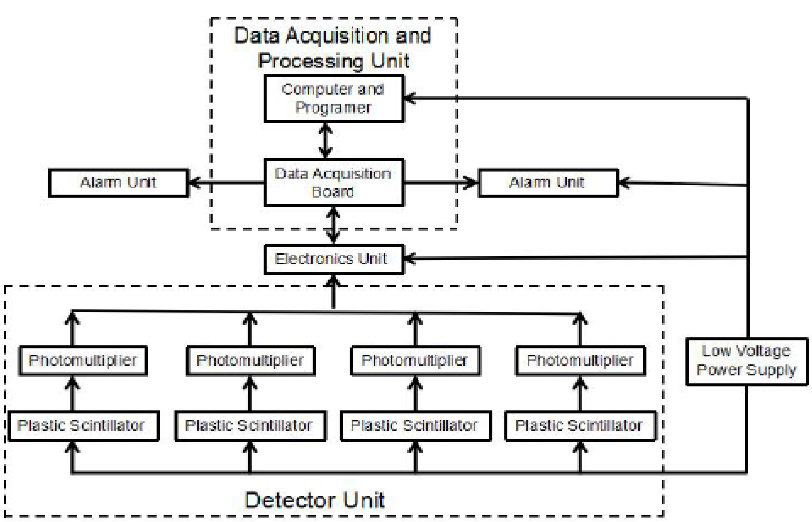

Fig. 1. Principle diagram of high sensitive detection system

PS2 (T2.1-0733)

New Ag-activated phosphate glass material for PRL dosimeter

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Current Ag-activated phosphate glass for dosimeter named FD-7[1] is widely used as one of the most reliable dosimeters in the world. The luminescent phenomenon is well known called RadioPhotoLuminescence (RPL). Especially, it has some notable properties the following: repeatable readout with the same intensity, less fading effect and reusable. We have achieved improving weathering resistance of phosphate glass in previous investigation[2]. The purpose of this study is to evaluate a new glass dosimeter with high weathering resistance that made by applying our technique.

Glass samples including the current glass (FD-7) were prepared the following procedure. Some raw materials such as Na_3PO_4, $AlPO_4$ and Ag_2O were mixed and melted at 1200°C in electric furnace for 3 h. The melted glasses were cooled rapidly into carbon mold. The glass samples were cut and polished. Dosed samples were prepared 0.01 to 3 Gy in varied exposure time using ^{137}Cs radiation source. Luminescent intensity was measured by photon counting method using UV-laser at 355 nm and photo multiplier tube.

Relationship between radiation dose and luminescent intensity is shown in Fig.1. The new developed glass dosimeter has high luminescence responsiveness as a function of radiation dose. Furthermore, change of repetition intensity in read-out and fading properties for dosed sample did not observed between the developed glass and the current glass. Weathering resistance property of the developed glass was increased about two or three times comparing to the current glass.

The improvement of weathering resistance will have the following two effects. It will bring about a decrease in the unit cost of radiation monitoring per time as more reuse times can be offered, and it can be used even in humid areas near the equator and in harsh outdoor environments that could not be used before. Therefore, the use of the developed glass with high weathering resistance would be able to make a dosimeter with an extended the range of use while maintaining the excellent reliability characteristics of the current glass.

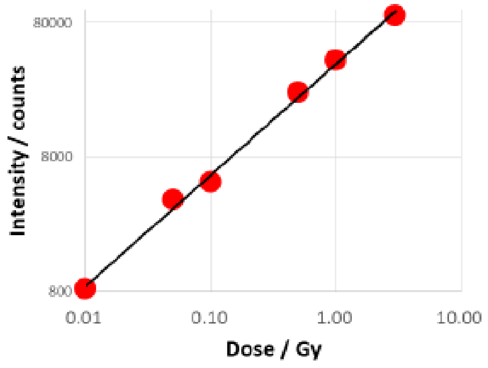

Fig. 1. Relationship between radiation dose and luminescent intensity.

Keywords: Glass dosimeter, RPL, High weathering resistance

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PS2 (T2.1-0745)

Screening for ^{137}Cs Body Burden Due to the Chernobyl Accident in Korosten city, Zhitomir, Ukraine: 2009-2018

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Chernobyl Nuclear Power Plant (CNPP) accident is the largest nuclear accident in human history that led to adverse health effects (thyroid cancer) and long-lasting radiation exposure. A huge amount of radioactive fallouts ended up with the high external and internal source of radiation exposure for people around CNPP in Russia, Belarus and Ukraine. Among various released radioactive materials from CNPP, ^{137}Cs was one of the major in terms of longevity of half-life (30 years) and capability to move around (transfer), not only on the ground but also from soil to plants and then into the human body through the intake of locally produced and wild forest products. These characteristics of ^{137}Cs have led to external and internal exposures of people living in in the contaminated areas during decades. Therefore, even 33 years after the accident, people residing around the CNPP continue receiving an inevitable chronic internal exposure due to ^{137}Cs that raises a permanent concern on possible health effects and contributes to maintain a high level of anxiety among the population.

Despite the numerous studies on the health effects resulting from internal exposure to low-dose radiation, scientists and experts have not yet reached a single final position on the issue. Considering this situation, it is essential to continue the monitoring of internal exposure and the health of the population to improve radiation protection and also identify potential health effects. Having access to the data of measurements carried out by Whole Body Counter (WBC) during the recent 10 years thanks to a collaboration with Zhitomir Inter-Area Medical Diagnostic Center (Center), we investigated and analyzed the Internal Exposure Dose (IED) of 110 thousand people living in contaminated Zhitomir region, Ukraine. This affected area is situated to the southwest of Chernobyl and experienced significant fallout of ^{137}Cs from CNPP.

Our aim is to examine the current general situation of IED in the residents considering the findings of recent studies and also to analyze their trend over the last 10 years. Correspondingly, our main focus is the analyze the evolution of the IED of all participants who underwent WBC screening in Center from 2008 to 2018.

Our study shows the presence of detectable internal doses in a substantial part of residents even in recent years. Although the ^{137}Cs concentration level in residents reduced progressively with time and is generally low, there are always people with a fairly high level of internal contamination for several years. The level of concentration and the frequency of people with internal dose were relatively high during the 2009-2012 period. This situation needs to be studied more in detail to identify the driving factors of the contamination and to make a more precise conclusion concerning the possible actions to improve it.

***Keywords**: Chernobyl Nuclear Power Plant accident, Internal exposure dose, ^{137}Cs*

PS2 (T2.1-0759)

Dose reassessment in TL dosimetry by using the phototransferred thermoluminescence (PTTL) phenomenon to the high sensitive LiF: Mg,Cu,P detectors used in environmental and eye dosemeters

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Thermoluminescent (TL) dosimetry is one of the most commonly used and spread method of dose measurements of ionizing radiation. The main and most inconvenient disadvantage of the TL method is an erasure of the TL signal on detector readout. Dose reassessment as post-readout is impossible in principle. While in film, OSL, and RPL dosimetry dose reassessment is well known and standard method since many years, in TL it was not possible to reassess the measured dose. This shortcoming can be eliminated by applying ultraviolet (UV) radiation. After the first readout, the same detector can be subjected to UV exposure and then read once again. This method of dose reassessment is based on the phototransferred thermoluminescence (PTTL) phenomenon and was applied to reassess doses in LiF: Mg,Ti (MTS-N) detectors used in individual and extremity dosimetry at Laboratory of Individual and Environmental Dosimetry (LADIS) IFJ PAN.

For environmental and eye lens dose measurements in LADIS laboratory, high sensitive LiF: Mg,Cu,P (MCP-N) detectors are being used. In the next step, the PTTL method was applied to high sensitive MCP-N detectors from routinely read dosemeters.

Standard MCP-N sintered detectors (4.5 mm diameter and 0.9 mm thickness used in environmental and eye lens dosemeters) were applied. Some of them have been used in routine measurements for 16 years long. The TL detectors were read in automatic RE2000 (Mirion Technologies) readers. After readout, the PTTL method was applied to dose reassessment for MCP detectors. Detectors were subjected to UV radiation and read once again. The response of MCP-N detectors to UV radiation was significantly lower than MTS-N detectors. The influence of UV wavelengths, UV irradiation conditions and the second readout parameters on the PTTL signal of MCP-N detectors and the possibility of dose reassessment were checked. Not reproducible background response, as well as different contributions of background in PTTL signal, caused a problem in the evaluation of small doses.

***Keywords**: thermoluminescence, dose reassessment, high sensitive LiF: Mg,Cu,P detectors*

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PS2 (T2.1-0761)

The analysis of different chromosomal aberrations in case of criticality accident

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Background

An accurate estimation of the absorbed dose is a key element for the evaluation of the risks associated with an exposure to ionizing radiation. However, depending on the radiation quality, a given absorbed dose could lead to different biological effects, the most prominent one being the level of DNA damage it can cause. Nowadays, the analysis of chromosome aberrations (CA) is considered the gold standard biological indicator of these effects as the complexity of CA can be correlated to the radiation quality. A criticality accident or use of atomic nuke could lead to people exposures to mixed gamma and neutrons radiation field, and could challenge the accuracy of the biological dose reconstruction based on CA.

Aim

In order to evaluate the level of CA complexity in case of exposure with a neutron component, we have performed exposure of human blood samples to mixed radiation field neutron fields of a reactor facility (Caliban reactor, CEA Valduc, France) similar to those generated during criticality accidents.

Methods

The comparison of CA complexity have been achieved using Multicolor Fluorescence In Situ Hybridization (M-FISH) of chromosome spreads obtained from blood samples exposed to 3 different radiation conditions: mixed field in steady state or pulsed mode (Caliban), and 4 MV X-rays (LINAC, IRSN) at the same doses.

Results/Conclusions

As expected, more complex aberrations were observed in the peripheral lymphocytes exposed to neutrons compared to X-rays. The number of breaks per cell between X-rays and neutrons exposure is significantly different. Interestingly, we have also measured a higher rate of unrepaired chromosomal fragments, leading to the hypothesis of a delay or impairment in DNA damage repair processes in this case.

Poster Session T2

PS2 (T2.1-0779)

Comparative study of natural radionuclides measurements and radiation dose assessment using vehicle-borne and laboratory gamma-ray spectroscopy techniques in Thailand

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In this study, vehicle-borne and laboratory gamma-ray spectroscopy techniques were compared to estimate the activity concentration of natural radionuclide in Thailand soil. The activity concentrations of ^{238}U (^{226}Ra), ^{232}Th (^{228}Ra) and ^{40}K in soil from the 6 regions of Thailand (77 points) were simultaneously measured with in-situ portable NaI(Tl) (a 3-in × 3-in scintillation spectrometer). In parallel, soil samples collected from these sites were analyzed with the laboratory gamma-ray spectrometry technique (HPGe). A good correlation was observed between both in-situ and laboratory analysis. The absorbed dose rate in air due to ^{238}U, ^{232}Th and ^{40}K in soils was calculated using the Beck's Formula and the results were compared with measured values obtained by vehicle-borne at 1 m height above ground. The results of the calculated and measured dose rate show a good correlation of R^2=0.79. Moreover, ^{232}Th was found to be the main contributor to the absorbed dose rate in air at 1 m from surface soil from outdoor terrestrial radiation in the study areas with a mean value of 31%.

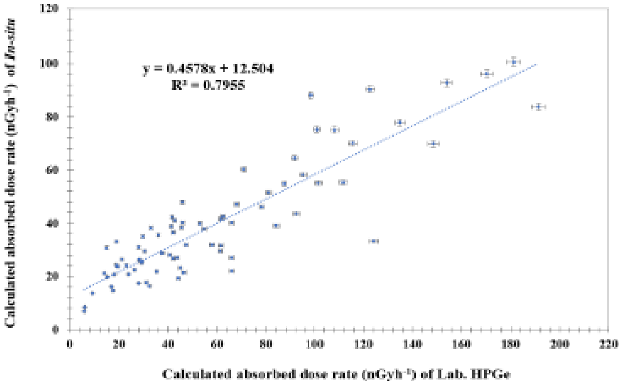

Fig. 1. The correlation between calculated absorbed dose rate with *in-situ* NaI(Tl) and laboratory HPGe

Keywords: Vehicle-borne, laboratory gamma-ray spectroscopy, Thailand

ACKNOWLEDGMENTS

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PS2 (T2.1-0788)

Establishing National Biodosimetry Laboratory for the Assessment of Accidental Radiation Exposure in Saudi Arabia

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Professionals working in the management of nuclear and radiological accidents are usually confronted with an array of complex issues when reacting to an emergency situation, regardless of whether they arise from disaster, negligence or deliberate act. An important medical aspect is the estimation of the biological radiation dose received by the victims. Therefore, founding of a biodosimetry laboratory is important for the estimation of absorbed dose in exposed individuals and population. The aim of this study was to establish the national biodosimetry service laboratory in the country. The main cytogenetic chromosomal aberrations assay (DCA), a proven, ISO and IAEA standardized technique for calculating medically relevant radiation doses (IAEA 2011, Al-Hadyan et al. 2014). Other ongoing biodosimetry assays include micronuclei and gamma-H2AX.

Peripheral blood samples were collected from 10 healthy volunteers of both gender. Aliquots of 2-ml whole blood were irradiated with doses between 0 and 5 Gy of 320 keV X-rays and incubated at 37C for 2 hours. Then, they were mixed with 18-ml of complete fresh culture medium, and lymphocytes were Phytohemagglutinin (PHA) stimulated, Colcemid division-arrested and cytogenetic slides were Giemsa-stained. The Metafer system was used for automatic and also assisted metaphase finding and scoring. DCA yields were fit to the linear-quadratic model of dose–response curve, according to the IAEA EPR-Biodosimetry-2011 report. We have successfully established the DCA dose-response curve, pre-required to estimate doses received. The linear and quadratic functions of the dose-response curve were in the range of those described in other populations. Although the automated scoring over-and-under estimated DCA at low (< 1 Gy) and high (> 2 Gy) doses, it showed potential for use in triage mode to segregate between victims with potential risk to develop acute radiotoxicity syndromes. The established standard calibration curve was used to estimate radiation doses received in real cases with suspected radiation exposures. The national biodosimetry laboratory has formed networks with international BioDoseNet in cooperation with IAEA/WHO.

In conclusion, we have established biodosimetry laboratory and produced a national DCA dose-response curve to help the nation's ability to respond to sporadic and mass radiation casualty incidents. The cytogenetic method can provide triage capability for rapid stratification of patients who need more specialized medical care and alleviate public concerns about the health effects of possible radiological exposures.

Keywords: Biodosimetry, Dicentric Chromosome Aberration Assay (DCA), Radiation Emergency Preparedness.

ACKNOWLEDGMENTS

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PS2 (T2.1-0805)

Extremity exposure with Tc-99m labelled radiopharmaceuticals in a nuclear medicine department

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Introduction: Nuclear medicine uses gamma and beta or alpha emitting radiopharmaceuticals for imaging and therapy respectively. Despite notable patient benefits, staff preparing and administering radiopharmaceuticals receive high radiation dose on their hands, in particular on their fingers, raising the risk of cancer induction. The study estimated radiation dose on fingers of radiographers who prepared and administered technetium-99m labelled radiopharmaceuticals.

Methods and Materials: Radiation dose received on fingers of radiographers who prepared and administered technetium-99m labelled radiopharmaceuticals were estimated using thermoluminescent dosimeters (TLDs). The radiation dose was measured at the bases and fingertips of the index and ring fingers of both hands of each of the participating radiographers, during a week-long study. Adhesive tape kept TLDs in fixed positions while radiographers worked with radiopharmaceuticals. Radiographers wore gloves to avoid contaminating the TLDs. The radiation dose absorbed by the TLDs were read with a TLD Reader with WinREMS™ (Model 3500).

Results: A total of five radiographers participated in the study, handling activity ranging from 78.20 MBq to 132.78 MBq per week. The dominant hands were the most exposed. A radiographer who handled 132.78 MBq per week received the highest dose (5.52 mSv) while the one handling 78.02 MBq received 0.71 mSv on the fingertips.

Conclusion: The finger tips and the bases of both the index and ring fingers of the dominant hands received higher radiation doses during preparation and administration of radiopharmaceuticals. However, the exposure of the radiographer who handled the most activity was unlikely to exceed the annual limit of 500 mSv assuming that the radiographer continued with the routine duty throughout the year.

Keywords: Finger doses, nuclear medicine, technetium-99m labelled radiopharmaceuticals, thermoluminescent dosimeters.

***Keywords**: Finger radiation dose, nuclear medicine, thermoluminescent dosimeters*

ACKNOWLEDGMENTS

We acknowledge with thanks the Department of Nuclear Medicine staff at Dr George Mukhari Academic Hospital in South Africa, who made it possible for this study to be successfully completed.

PS2 (T2.1-0809)

An Algorithm to Assess Internal Exposure Dose for Workers in Decommissioning Nuclear Power Plant

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To assess internal exposure dose for workers, a long half-life radioactive crud which remains in system and/or pipe is used for source term. Remaining radioactive crud inside the system, structure and components(SSCs) can be dispersed in the working area by cutting process. Afterward, dispersed radioactive materials can be inhaled into the human body through breathing.
For conservative assessment, long half-life radioactive materials are used for calculation. Since the longer half-life, the greater effect on the human body.
Schedule No. 40 stainless pipes are used for calculation. Each pipes with four different diameters are used since there are various kind of pipes which have different sizes in the nuclear power plant. For conservative calculations, assumptions which workers do not wear radiation protective equipment, and do not operate heating, ventilation, and & air conditioning(HVAC) for removing floating radioactive materials are applied.
Since decontamination will be performed on several systems prior to decommissioning, decontamination factor(DF) is applied to the internal exposure calculation. Floating rate, respiration rate and diffusion factor are used to calculate how much radioactive materials are inhaled in the human body through respiration[1-2].
Calculation results of internal exposure dose are shown in Table 1.

Table 1. Calculation results of internal exposure dose

Diameter of pipe	DF 50		DF 100(mSv)	
	Primary(mSv)	Secondary(mSv)	Primary(mSv)	Secondary(mSv)
1 in	5.2E-05	5.5E-06	2.6E-05	2.6E-06
2 in	1.0E-04	1.0E-05	5.2E-05	5.2 E-06
3 in	1.5E-04	1.5 E-05	7.7E-05	7.7 E-06
4 in	2.0E-04	2.0 E-05	1.0E-04	1.0 E-05

Internal exposure dose varies according to the diameter of pipe and DF. As the diameter of pipe increases, so does internal exposure dose. In addition, the internal exposure dose of the primary system is greater than that of the secondary system. Comparing calculation result of DF 50 and DF 100, the internal exposure dose is lower when DF is 100.
The bigger diameter of pipe, the higher the internal exposure dose since there are more radioactive materials inside. The higher DF, the lower the internal exposure dose since more radioactive crud is removed. In addition, internal exposure dose for the primary system is higher since the primary system is more contaminated by radiation materials than secondary system. Calculated internal exposure dose of workers by using this methodology will be used for the Kori-1 FDP.

Keywords: NPP decommissioning, Workers' exposure dose, Collective dose

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PS2 (T2.1-0810)

Software Development to Assess Work Exposure Dose of RVI Decommissioning Process

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Various radiological protection contents about decommissioning will include in the Final Decommissioning Plan(FDP), and quantitatively calculated exposure dose to workers by decommissioning operations should be described in Radiation Protection part[1-2]. Since the Kori-1 nuclear power plant was permanently shut down for the first time in South Korea, it is necessary to develop related methodologies and software, which could be utilize in other nuclear power plant in the future.

Exposure dose from Reactor Vessel Internal(RVI) decommissioning process is expected to have the greatest effect for workers' exposure dose. Therefore, RVI was decommissioned underwater at overseas nuclear power plants, and RVI of Kori-1 nuclear power plant will also be decommissioned underwater. If decommissioning is performed underwater, the effects of radioactive material can be minimized. In addition, the radiation shielding effect of water is much better than that of air.

Since RVI decommissioning will be performed underwater, workers will be exposed by scattered radiation due to the shielding material such as water and concrete. In order to consider effect of scattered radiation for workers, probabilistic simulation tool should be used. For this reason, Monte Carlo N-Particle Transport Code(MCNP) is used to calculate exposure dose of workers for RVI decommissioning process. The results of radioactive evaluation of Reactor Vessel(RV) and RVI are used as source term of MCNP. In addition, for the most conservative assessment, radiation sources inside RV and RVI are existed. The distance between the top of the RV and the surface of the refueling cavity is assumed to be as close as possible.

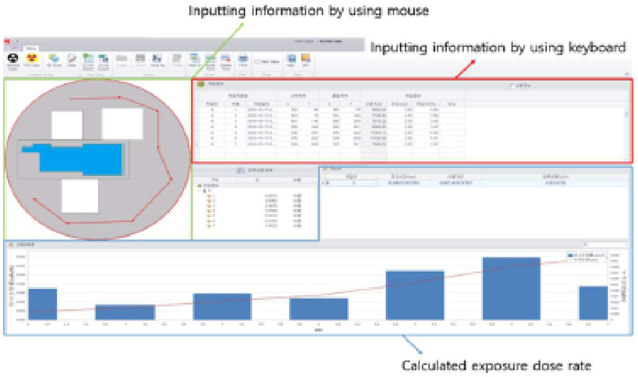

Fig. 1. Developed software to calculate exposure dose for RVI decommissioning process

As shown in Fig. 1, exposure dose by phase of decommissioning can be calculated based on work line. Work line for workers can be created by typing on a keyboard or mouse, and work line can be identified by red line in the software. Since the speed of movement and stop working can be applied, exposure dose for workers can be calculated more accurately.

Keywords: NPP decommissioning, Exposure dose for RVI decommissioning, MCNP

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PS2 (T2.1-0811)

A Methodology for Calculating External Exposure Dose of Workers in Decommissioning Nuclear Power Plant

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In 2017, Kori-1 was permanently shut down and the Final Decommissioning Plan(FDP) for Kori-1 should be submitted to the regulation authority in accordance with relevant regulations. Since workers are expected to perform various decommissioning operations depending on the stage of decommissioning, it is essential to predict exposure dose of workers that inevitably occurs in this process[1-2].
The exposure dose of workers in the decommissioning work be managed by collective dose. The calculating collective dose was also applied in the decommissioning of overseas nuclear power plants, and this method is useful in calculating exposure doses for entire nuclear power plant area. Furthermore, optimized number of workers can be calculated by using predicted collective dose.
Collective dose can be calculated using work time, number of workers, and spatial radiation dose rate. Work time and number of workers from Unit Cost Factor(UCF) and spatial radiation dose rate from Kori Unit 1 are used for calculation. For efficient collective dose calculations, nuclear power plant is divided into various work areas and the collective dose is calculated for all work areas.
The workers are divided into general workers and technical workers. General workers perform common tasks, such as transport material and surveillance. Technical workers perform decontamination and cutting process near the contaminated material. The general works are assumed to be performed throughout the work area, and the technical work is considered to be conducted at a distance of 1m from the contaminated material.
Various MCNP simulations were conducted to compare the exposure dose of general workers with that of technical workers when working in the same work area. Results of MCNP simulations are shown in Table 1.

Table 1. Exposure dose for general workers and technical workers when working in the same work area

A: Exposure dose for general workers (mGy)	B : Exposure dose for technical workers (mGy)	A/B Ratio (%)
8.82E-08	8.83E-07	10.0
9.16E-08	8.74E-07	10.5
5.13E-08	4.40E-07	11.7
9.07E-08	8.07E-07	11.2

Based on result of Table 1, exposure dose for general workers is about 10% that of technical workers. The Decontamination Factor(DF) is conservatively assumed to be 10. The major nuclide causing radiation exposure to workers during the immediate decommissioning of nuclear power plant is Co-60[2]. The half-life of Co-60 is 5.2714 years and decommissioning may begin 10 years after permanent shutdown. For this reason, the collective dose is divided by 4. The total collective dose is calculated by using calibrated work time, number of workers, weighted spatial radiation dose rate, and influence of half-life.

Keywords: NPP decommissioning, Workers' external exposure dose, Final Decommissioning Plan(FDP)

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PS2 (T2.1-0812)

Verification of Workers' External Exposure Dose Assessment Algorithms

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In the Final Decommissioning Plan(FDP), exposure dose of workers during decommissioning process and radiation protection should be included[1]. To calculate exposure dose for workers, it is necessary to obtain detailed information of source which causing exposure. However, during the decommissioning stage on nuclear power plant, the work area will change continuously unlike normal operation. For this reason, it is difficult to obtain accurate source information, an algorithm based on Electric Power Research Institute(EPRI) was developed to calculate workers' exposure dose by using limited information[2].

Input information of developed algorithm is position of source and spatial radiation dose rate only. Even if spatial radiation dose rate of the source is not known, it is possible to calculate exposure dose for workers based on developed algorithm. In this research, the developed algorithm has been validated in various method to apply in the decommissioning nuclear power plant.

For the EPRI algorithm, warning statement is shown to the user if error range is 50 to 100%, and a rejected statement is presented to user if error range is more than 100%. Therefore, it is a reliable result if error range is between 0 to 50%. The same error condition was used since the algorithm was developed based on EPRI algorithm.

To verify developed algorithm, Monte Carlo N-Particle Transport Code(MCNP) which utilized widely was used. The simulation conditions are as follows. One source was used and consists of Co-60 30%, Co-58 50% and Ag-110m 20%. There were two measuring points to calculate exposure dose and one shielding wall was existed. There are 4 points to calculate the radiation dose rate.

In addition, developed algorithm was also verified by experiments. Radiation dose rates were measured at 430mm, 1330mm, 2230mm, 3130mm and 4030mm respectively, and measured radiation dose rates compared with calculated radiation dose rate by using developed algorithm.

The radiation dose rates estimated by the algorithm and those calculated results from MCNP are shown in Table 1.

Table 1. Radiation dose rates calculated by developed algorithm and MCNP

Point	Radiation dose rate by developed algorithm (mSv/hr)	Radiation dose rate by MCNP (mSv/hr)	Relative error (%)
A	316.0	315.9	0.03
B	142.0	143.4	0.99
C	78.3	79.0	0.89
D	35.1	35.9	2.28

Developed algorithm data compared to MCNP were in the range of 0.03~2.28% relative error. In other words, developed algorithm represent high accuracy not only simulation but also experiment.

Keywords: NPP Decommissioning, Exposure dose calculation algorithm, MCNP

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PS2 (T2.1-0813)

A Methodology of Calculating Exposure Dose for RV, RVI Decommissioning Work in Nuclear Power Plant - a Monte Carlo Study

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In 2017, the first commercial nuclear power plant in Korea, Kori-1 was permanently shut down. Based on the nuclear safety act, the Final Decommissioning Plan(FDP) for Kori-1 should be submitted to the regulation authority. In the FDP, exposure dose of workers during decommissioning stage and radiation protection should be included[1].
Reactor Vessel Internal(RVI) decommissioning work among the various decommissioning works is expected with high exposure dose. Therefore, detailed and accurate assessments of exposure dose for workers are needed to reduce workers' exposure dose. In case of overseas decommissioning nuclear power plants, cutting process of RVI is conducted underwater to reduce exposure dose[2].
An algorithm for calculating exposure dose during RVI decommissioning process is developed based on Monte Carlo N-Particle Transport Code(MCNP). Since MCNP is a probabilistic assessment method, the results are highly reliable. In addition, scattered radiation due to the shielded water is also reflected accurately in the results.
Various results can be obtained through tally in MCNP. Using mesh tally, the desired area can be divided into lattice patterns and results can be obtained according to each lattice. Reactor Vessel(RV) and RVI in the water is used for source term, and external are of spent fuel pool where workers can move during RVI decommissioning process is used for desired area in mesh tally. The exposure dose is calculated by drawing the workers' work line in the area and summing the results of mesh tally which overlapped with the workers' work line.

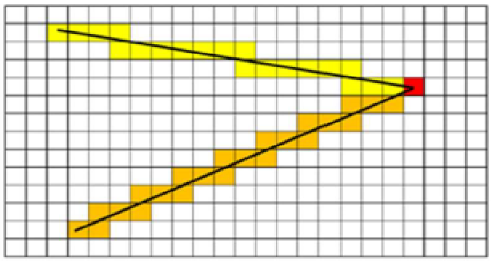

Fig. 1. Algorithm for calculating exposure dose for workers in RVI decommissioning work

In Fig. 1, black line is the work line, yellow and orange squares are the areas which overlapped with the work line and mesh tally results. Red square is the area where workers stay. For red square, the exposure dose is calculated by multiplying the result of mesh tally with the work time.
Based on this methodology, exposure dose for workers during RVI decommissioning will be calculated accurately. Furthermore, safe work plan will be established and optimization of exposure dose for workers will be possible.

Keywords: *NPP decommissioning, Workers' exposure dose, MCNP*

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PS2 (T2.1-0815)

Calculation of equivalent dose to the lens of the eye for patients during head IVR procedures by using detailed eye model

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Objective In Interventional Radiology (IVR), there are reports of problems associated with prolonged fluoroscopy. In recent years, measures have been taken to reduce the exposure by devices, therefore patient doses have decreased. However, with head IVR procedure, the development of patient's cataract cannot be ignored if the lens of the eye enters an irradiation field. So, we evaluated the equivalent dose to the lens for the patients during head IVR procedures using a Monte Carlo simulation code "PHITS" (Particle and Heavy Ion Transport code Systems) and the MIRD phantom incorporated the detailed eye model.

Methods Using the MIRD phantom as a patient, a field of head IVR procedure was simulated and equivalent dose to the lens of the eye for patients was calculated. The calculation conditions are as follows: assumed procedure is intracephalic aneurysmal embolization, the irradiation field is 8 inches × 8 inches(Posterior-Anterior: PA direction) and 10 inches × 10 inches(Left-Right: LR direction), the source position is 65 cm from the back of head of the MIRD phantom, the tube voltage is 93 kV(PA) and 92 kV(LR) and the shape of the X-ray is a cone beam. Calculating the ratio of calculated $H_p(0.07)$ and measured $H_p(0.07)$ as conversion factor, then multiplying calculated equivalent dose to the lens by it. So, estimating the equivalent dose to the lens of the eye for patients under the actual irradiation condition.

Results The equivalent dose of LR direction (maximum value) was about 17 times of PA direction (average value). In the LR direction, the equivalent dose of the right eye was about one quarter of the left eye. In the PA direction, X-rays pass through the head, so the equivalent dose of the lens is reduced. In the LR direction, there is a high possibility that X-rays are directly incident on the lens.

Conclusion We developed the calculation method that can estimate the dose to the lens for patients considering the angle of the X-ray tube and the fluoroscopic time without measuring the dose of patients during head IVR procedures.

***Keywords**: Equivalent dose to the lens of the eye, Head IVR procedure, ICRP detailed eye model*

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PS2 (T2.1-0818)

Development of Korean-specific thyroid uptake rate of radioiodine for internal dosimetry

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Internal dose from radioiodine intake, which is the main contribution to the early phase dose of nuclear accident, is generally estimated using ICRP biokinetic model and corresponding biokinetic parameters that represent physiological characteristics for reference person. However, different dietary iodine levels from that of the reference person can impact on the physiological behavior of radioiodine (e.g., thyroid uptake rate of radioiodine) and thus lead to significantly different thyroid dose estimate. Therefore, for South Korea as one of the iodine-rich countries, Korean-specific thyroid uptake rate of radioiodine should be considered to more accurately estimate the thyroid dose.

In this study, the thyroid uptake rate (i.e., blood-to-thyroid rate (d^{-1}) in iodine biokinetic model) was newly developed on the basis of 24-h thyroid uptake data obtained from twenty-eight Korean euthyroid adult males [1]. To determine the thyroid uptake rate best describing the thyroid uptake of Korean subjects, the new ICRP iodine biokinetic model was repeatedly calculated with varying the blood-to-thyroid rate and the calculation results were statistically compared with the 24-h thyroid uptake measurement data using chi-square test. After then, the thyroid equivalent dose coefficient was also calculated using the determined thyroid uptake rate for comparison with that of ICRP reference. As the results, the Korean-specific thyroid uptake rate was determined as 5.04 d^{-1} (p-value > 0.05), which was significantly lower than ICRP reference value (7.26 d^{-1}). In addition, this Korean-specific value produced the thyroid dose coefficients of 2.8×10^{-7} Sv/Bq, which was 78 % of ICRP reference value (Table 1). The results in this study show that for Korean less radioiodine would be absorbed to thyroid and the thyroid would be less irradiated than ICRP reference person after equal radioiodine intake.

Table 1. Comparison between Korean-specific and ICRP reference values

	Korean-specific	ICRP [2]	Korean-specific /ICRP
Thyroid uptake rate (d^{-1})	5.04	7.26	0.69
Thyroid equivalent dose coefficient (Sv/Bq)	2.8×10^{-7}	3.6×10^{-7}	0.78

Keywords: *Internal dose, Thyroid uptake, Radioiodine exposure*

ACKNOWLEDGMENTS

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PS2 (T2.1-0827)

Current Status of Regulatory Requirements and Technical Standards on Radiation Dose Limits for the Lens of the Eye

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To protect the lens of the eye in occupationally-exposed radiation workers, the International Committee on Radiological Protection (ICRP) defines the equivalent dose of the lens of the eye as a protection quantity and proposes an annual dose limit. ICRP Publication 103, issued in 2007, intended to lower the dose limit taking into account the results of epidemiological studies conducted since 1999. In 2012, ICRP Publications 103 and 118 lowered the threshold dose and the dose limit for the lens of the eye (Table 1). As ICRP Publication 60, the previous recommendation in 1990, is replaced by ICRP Publication 103, the European Union and the International Atomic Energy Agency (IAEA) published revised safety standards. On the other hand, U.S. Nuclear Regulatory Commission (U.S. NRC) did not apply ICRP Publication 103 to the 10 Code of Federal Regulation (CFR) Part 20 in response to opinions of stakeholders (nuclear and radiation-related industries). In Korea, the regulation of radiation protection is currently based on ICPR Publication 60. The Nuclear Safety and Security Commission (NSSC), the Korean regulatory body, conducted the research project to apply ICRP Publication 103 to the regulation and resulted in the draft of the occupational dose limit for the lens of the eye, the same as the ICRP dose limit.

Table 1. Comparison of the threshold doses and dose limits for workers in ICRP Publication

Classification		ICRP Publication 60	ICRP Publications 103 & 118
Threshold dose	Lenticular Opacity	Single exposure: 0.5 - 2 Gy Chronic exposure: 5 Gy	0.5 Sv (or 0.5 Gy)
	Cataract	Single exposure: 5 Gy Chronic exposure: 8 Gy	
Occupational dose limit	Lens of the eye	150 mSv y^{-1}	100 mSv 5y^{-1} [a] (50 mSv y^{-1})

[a] Occupational dose limits of 100 mSv over five years with no more than 50 mSv in a single year

The International Commission on Radiation Units and Measurements (ICRU) proposes to use a personal dose equivalent at a depth of 3 mm in body tissue as an operational quantity to measure the radiation exposure to the lens of the eye. Technical specifications for lens dosimeters are provided in the International Electrotechnical Commission (IEC). The International Organization for Standardization (ISO) provides the dose level of protection for the lens of the eye (6 mSv y^{-1}). Since the occupational dose limits were reduced by approximately 25 times from 500 mSv y^{-1} to an average 20 mSv y^{-1}, monitoring of dose to the lens will be an issue of radiation protection for workers, especially for isotope production and nuclear industry. This paper will be helpful for readers seeking information about regulatory requirements and technical standards on radiation exposure to the lens of the eye for workers.

Keywords: Lens of the Eye, Equivalent Dose, ICPR Publication 103

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PS2 (T2.1-0837)

Statistical analysis of measured calibration factors of numerous different dosemeters obtained over more than 10 years

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The Austrian legislation requires verification (similar to calibration) of all officially used active dosemeters for photon radiation on a regular basis. Calibration intervals of two calendar years are mandatory. In the accredited testing-, calibration- and verification-laboratory of the Seibersdorf GmbH calibration factors of more than 1000 different dosemeters were determined each year already since 2006. In most cases reference radiation qualities are used. This high number of data allows a detailed statistical analysis. Both a comparison of different dosemeter types as well as an evaluation of the stability of single instruments are given. The results are also referred to relevant international dosimetry standards (ISO, IEC).

These presented data allow a properly assessment of the uncertainty contribution of the instrument calibration of dosemeters to the overall uncertainty of the dose measurement.

Keywords: *calibration, uncertainty, dosemeter*

PS2 (T2.1-0872)

Fabrication of Plastic Scintillator for Measuring Low Energy β-emitter Radionuclide

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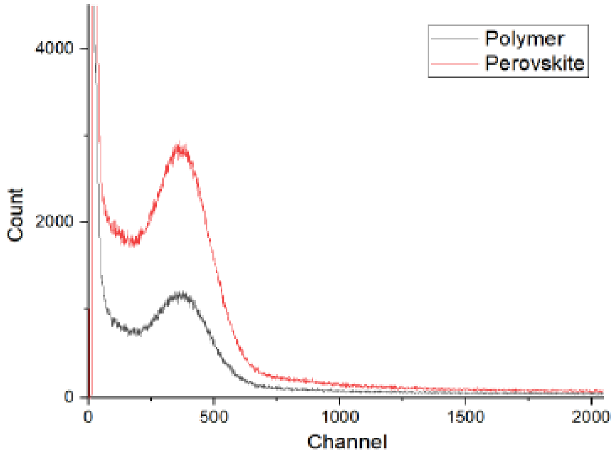
Radiation detection materials are widely used in many applications has led to extensive research on scintillators. Especially, halide perovskite materials emerge as promising candidates for radiation detectors because they were shown to be good photodetector nanomaterials due to the large light absorption factor, tunable bandgap, long carrier recombination lifetime, and large mobility [1]. In this study, we report a perovskite based scintillator was able to be fabricated and tested to measure low energy emitting isotopes through adding perovskite nanomaterial to organic scintillators to manufacture Organic-Inorganic Hybrid scintillators. The perovskite scintillator emits strong light from its particular wavelength through quantum confinement effect that can lead to high efficiency.
In this experiment, the perovskite nanomaterial was mixed with epoxy to manufacture plastic scintillator. To compare the efficiency of plastic scintillators, a radiation measurement experiment was conducted using the sealed point source of low Energy β-ray emitting Cl-36 [2,3].



It was shown that the efficiency of conventional plastic and perovskite added plastic was 5% and 14%, respectively. And the efficiency of perovskite added increased to about 3 times compared to conventional plastic scintillator. Thus, by comparing the efficiency of perovskite driven scintillator with current state-of-art radiation detectors, we showed the promising features and challenges of nanomaterials as promising radiation detectors.

Fig. 1. Radiation Spectrum of polymer(black) and perovskite added polymer(red), respectively

Keywords: Plastic Scintillator, Low Energy Beta, Nanomaterial, Perovskite

ACKNOWLEDGMENTS
This work was supported by a grant from the National Research Foundation of Korea(NRF), funded by the Korean government, Ministry of science and ICT.[No. 2017M2A8A5015143]

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PS2 (T2.1-0874)

Dose Assessment for Accident Scenarios During Use of Spent Resin Treatment Device

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In the heavy-water reactor, ion exchange resins are used to purify the liquid radioactive waste generated during operation and are stored in storage tanks. The resins stored in the storage tanks should be handled in accordance with future decommissioning plans. But the resins, which is an intermediate level waste, generated from the heavy-water reactor cannot be disposed of in the radioactive waste disposal facility because the concentration of ^{14}C exceeds its low level radioactive waste disposal standard in accordance with Nuclear Safety Act in Korea. For this purpose, a demonstration device with a daily processing capacity of 1 kg was developed to develop a commercialized resin treatment device that can remove ^{14}C from resin and lower it to low level waste [1]. In this study, an accident scenarios of a treatment device with a treatment capacity of 1 ton/day was established and dose assessment was performed.

Accident scenarios were set for the resin mixture separation device under pressure and the microwave device under pressure and heat as shown in Fig. 1. In the resin mixture separator, approximately 100 kg of resin mixture is processed per hour. It was assumed that shutoff valve between the inlet-separator connecting pipe is activated to prevent the outflow of additional spent resin mixture. It is also assumed that for an additional 1 minute, the worker wears a safety respirator equivalent to APF (Assigned Protection Factors) of 1,000 and performs the task of blocking the fracture surface of the separator. Thus 2.5 kg of the resin mixture is released, resulting in 0.272 mSv of internal exposure to worker. In the microwave device, about 80 kg of resin is treated per hour. It was assumed that the shutoff valve between the separator-microwave connection pipe was activated. It was also assumed that for an additional minute, the worker would block the fracture surface with safety respirator. Thus 2 kg of the resin is released, resulting in 0.214 mSv of internal exposure to worker. In both cases, it was confirmed that the external exposure dose does not exceed 20 mSv even when combined with each internal exposure dose. Therefore, it was confirmed that there is no problem in safety during the accident.

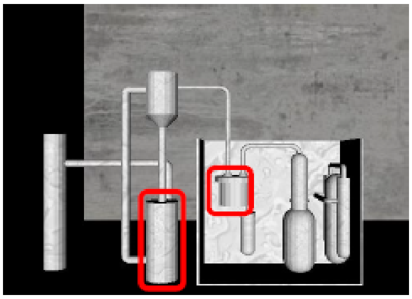

Fig. 1. Expected accident location of spent resin treatment device

Keywords: Spent resin, Dose assessment, Accident scenario

ACKNOWLEDGMENTS

This work was supported by the Korea Institute of Energy Technology Evaluation and Planning and the Ministry of Trade, Industry and Energy & the National Research Foundation of Korea (NRF)

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PS2 (T2.1-0878)

Dose assessment for remote and adjacent-work scenarios for the spent resin treatment facility

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Spent resin from the heavy-water reactors includes a variety of radionuclides that are classified as ILW (Intermediate Level Waste) containing ^{14}C nuclide with a high concentration of 8.06E+06 Bq/g. ILW according to Nuclear Safety Act in Korea cannot be disposed of in caves because it is not approved for cave disposal facilities, where the concentration of ^{14}C in this resin exceeds its tolerable limit (2.22E+05 Bq/g) for near-surface disposal of low level radioactive waste [1]. So, waste resin treatment is necessary to reduce the radioactivity concentration of C-14 below the low-level radioactive waste disposal criteria.

The 1 ton/day spent resin treatment facility separates the zeolite and activated carbon and deregulates it, ^{14}C of spent resin is desorbed through microwave, and separated spent resin (Cs-137, Co-60, Eu-152, etc.) is stored in spent resin storage part in the facility. In this study, dose assessment was conducted for the workers working in adjacent work and remote work. Remote work is considered to conservatively meet the annual dose limit (100 mSv for 5 years without exceeding 50 mSv per year) of the workers [2]. As shown in table 1, it is determined that the annual dose for the 8 hr/day work in remote work satisfies the dose limit.

Table 1. Comparison of annual dose between adjacent work and remote work

	Adjacent work (20 cm)	Remote work
Dose rate (mSv/hr)	3.60E-01	3.90E-03
Annual dose (1 hr/day)*	9.00E+01	9.75E-01
Annual dose (8 hr/day)*	7.20E+02	7.80E+00

*The worker is assumed to work 5 days for a week and 50 weeks for a year.

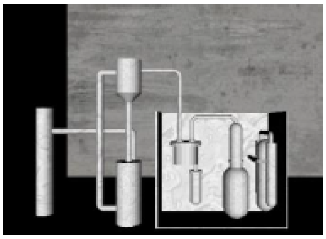

Fig. 1. 3D modeling of the 1 ton/day spent resin treatment facility

Keywords: Spent resin, External dose, ^{14}C

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PS2 (T2.1-0895)

Usefulness of Locally Constructed Phantoms for the Validation of Planned Radiation for Breast Cancer Radiotherapy Treatment

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GLOBOCAN estimates, indicate that 4645 new cases were diagnosed and 1871 death occurred due to breast cancer in Ghana in 2018; making it the commonest female cancer and a major public health problem. To ensure the facilities in Ghana implement quality control measures, this study was designed to determine and compare planned with actual doses delivered to the breast during treatment. This is to achieve this, the major limitation of the non-availability of phantoms was addressed by the construction of phantoms, using perspex and locally available materials that mimic organs of the female thoracic cavity. Based on scanned images, two phantoms were constructed.

Balloons, mango seed, cassava stick and candle were radiologically assessed and used as surrogates for the lung, heart, spinal cord and glandular tissue of the breast respectively. Higher photon energies from a ^{60}Co and LINAC machine were targeted at the left breast of a standard and the two constructed phantoms. EBT3 film dosimeter was used to measure absorbed doses to the breast and non-target organs.

The deviations of delivered doses from planned doses when the standard anthropomorphic phantom, constructed phantoms A and B were used, ranged as follows, -0.05 – 0.03 Gy; -0.08 – 0.01 Gy; -0.14 – 0.01 Gy respectively, when the radiation was delivered by a Cobolt-60 machine. When the radiation was delivered by a linear accelerator systems, the deviations were -0.05 – 0.03 Gy; -006 – 0.07 Gy; -0.06 – 0.04 Gy respectively. The left lung and spinal cord received the highest and lowest unintended dose, 0.74±0.04 Gy (Co-60) and 0.78±0.01 Gy (LINAC), and 0.03±0.02 Gy and 0.05±0.01 Gy respectively for phantom A.

The study has demonstrated that local materials are potentially useful for the construction of phantoms, which can be good substitutes for standard commercial phantoms in ensuring the safety of patients under-going radiotherapy treatment for breast cancer.

Keywords: *Breast Cancer, Phantoms, Radiotherapy*

ACKNOWLEDGMENTS

The authors would like to thank Sweden Ghana Medical Centre and National Centre for Radiotherapy and Nuclear Medicine, Korle Bu Teaching Hospital for sharing their facilities. Furthermore, we express our gratitude to the Department of Physics, University of Cape Coast and Ghana Education Trust Fund (GETFund).

PS2 (T2.1-0901)

Passive detector response to low energy radiation and influence of the exposure inherent conditions on dose measurement

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The occupational exposure dose monitoring is in the most part performed with passive detectors. The first reason for which the passive dosimeters are widely used is their property to cumulated small doses through a period without to loss the information. The passive detectors are mandatory worn in dosimeter badges with different design and equipped with different filters. The detector calibration is performed in a badge and for a conventional true value of dose the detector gives a response. It is possible that the badge does not influence the detector response as the response of the certain types of passive detectors are less influenced by the radiation energy. How and how much the dosimeter badge influences the accuracy of the detector response is well to know.

Generally, the occupational exposers work in safety conditions of radiation. In certain nuclear fields such as radioisotope production or nuclear medicine the workers record monthly small doses of radiation between 0.1 mSv and 1.0 mSv and rarely higher doses over 1.0 mSv. The sources of low energy are used in medicine for radiation therapy or diagnostic and in industry for non-destructive control.

What is the difference between the response of the free detector materials and these of the detector worn in a dosimeter badges in the low dose range and low energy exposure is presented in this work.

In this respect, TL and film detectors with or without badge have been exposed at different doses of low energy radiation and the response accuracy has been studied. The exposure to low energy was performed with X-Strahl generator and Am 241 standard source. The detector exposure was performed at different voltage between 40 kV and 150 kV and different doses on the 0.05 ± 10.00 mSv range. For an exposure to 40 kV and 1.00 mSv the film responses were 4.73 ± 0.07 odu (optical density unit) and 4.70 ± 0.005 odu in free conditions and in the FD-III-B type dosimeter badge, respectively. The film responses were 3.56 ± 0.09 in free conditions and 3.34 ± 0.005 in badge in the case of the exposure to 0.5 mSv. The optical density values and their standard deviation are higher in free conditions of exposure. On the other hand, the film dimensions measured in free condition were higher than those measured on the badge "window" and not only, other details related to the exposure geometry have to be fixed. How these shortcomings influence the dose assessment on free area of the detector are studied in this paper.

Keywords: *dose accuracy, detector, low energy*

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PS2 (T2.1-0906)

Dose inspection on radiation safety for the use of veterinary x-ray machines in Taiwan

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Radiation diagnosis technology has been used in animal for many years and become more and more popular in Taiwan. The number of veterinary hospital is growing rapidly in recent two decades in Taiwan. In most cases, the helper, usually the pet breeders, needs to accompany their pet inside the X-ray room during the X-ray inspection. Radiation doses and risks of the helper thereby are highly concerned The purpose of this study is to investigate the exposed doses of the helpers inside the X-ray inspection room during veterinary diagnostic inspection procedures, and also investigates the characteristics such as radiation leakage and the total filtration of X-ray tubes. A plastic scintillation survey meter (Atomtex AT1121) was used to measure the doses at the position of the helper inside the X-ray room and also used to measure the leakage of X-ray tube. Besides, the pen dosimeters with different thickness of Al shielding rings were used to assess the total filtration of X-ray tubes. There were 147 veterinary hospitals inspected on-site during 2017 to 2019. By means of setting the survey meter at the position of the helper's body which is assumed at a distance of 50 cm from X-ray field center, and considering the conditions of wearing with/without lead apron respectively, the ambient dose is measured by the survey meter and then transfer into effective dose by considering the conversion factors. The effective doses at the helper's positions in cases of with and without wearing lead apron during the x-ray inspection are investigated and reported in this study. In addition to assess the doses of the helper, this study also indicated the results of qualities of diagnostic X-ray equipment used in veterinary in Taiwan.

PS2 (T2.1-0908)

Development of a High throughput Biodosimetry Laboratory in a Reference Hospital for Medical, Occupational Applications and Response to Radiological Emergencies

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Biodosimetry consists of the use of physiological, chemical or biological markers of human tissue exposure to ionizing radiation for the purpose of dose reconstructing appropriate to individuals or population groups. The International Organization for Standardization (ISO) established a common set of rules, with the publication in 2008 of the 21243 Standard and then in 2014 the 19238 Standard. Both of these were related to Biodosimetry. The International Atomic Energy Organization (IAEA) produced two technical manuals related to Biodosimetry in 2001 and 2011 (1,2) .

Automated cytogenetic dose assessment methods have become imperative for efficient preparedness in response to radiation events involving mass casualties. This automatization increases throughput since the techniques involved are laborious and time-consuming. Automatization also improves quality control and safety assurance for laboratory personnel. In 2019, the University Hospital La Fe obtained a FEDER EC grant of 600,000 Euros for a project to automate all the steps involved in the Biodosimetry process. Cytogenetic laboratory automation includes metaphase harvesters, a spreader and slide aging system, to automate sample preparation, metaphase finding and image capture, automation of the dicentric assays, and dicentric scoring software.

Biodosimetry is an accepted procedure for use in radiological and/or nuclear emergencies, and in 2001, the first International Networks were established in response to radiological and/or nuclear emergencies. We will outline the experience of RENEB (The European Network). In addition to the application of Biodosimetry its use needs to be optimized and performed in medical applications for patients involving radiotherapy (RT) and radiological treatment. In these two fields Biodosimetry investigation is used for dose reconstruction in patients after the first RT fraction or radionuclide administration. Measurements of biomarkers in RT and diagnostic radiology patients during the course of their procedures endeavors to generate in vivo dose response curves related to accidental overexposures in radiological practice, testing and validation of new biomarkers of radiation exposure, assessment and prognoses of the acute or late radiation toxicity in RT patients from in vivo yields of radiation biomarkers (3).

Another important field of application s using Biodosimetry techniques for workers involved in diagnostic and/or treatment procedures using ionizing radiation as outlined in the "Royal Decree783/2001of 6th July of the Spanish Health Protection Regulation (RD 783/2001)". Our proposal is to estimate the absorbed dose to complement the physical Dosimetry in workers involved in treatment are frequently exposed to protracted and fractionated low doses of ionizing radiation, which extend during all their professional activities. These exposures to direct and scattered radiation can result in, deterministic effects (radiodermitis, skin aging, cataracts, telangiectasia in the nasal region, vasocellular epiteliom s, hand depilation) and/or stochastic effects

PS2 (T2.1-0930)

Evaluation of Irradiation Time Error for Calibration of Neutron Personal dosimeter

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In order to calibrate the personal dosimeter, the dosimeter is exposed to the reference radiation field for a certain time and the calibration factor is calculated by dividing reference value by the indicated value. However, the irradiation time error between the setting time derived by numerical model of personal dose equivalent and the actual irradiation time could occur by the resolution of timer and the characteristics of source transport system of the irradiator. And it could result in overexposure than expected. Therefore the irradiation time error should be estimated and compensated or reflected on the irradiation time uncertainty. ^{252}Cf neutron irradiator installed at KHNP CRI was designed that neutron source is withdrawn from the shielding container by the air pressure and returned in a free-fall manner. Recently, the source holder has been changed as a part of maintenance of irradiator. It could be possible to affect the source transit time. This study was conducted in order to figure out the effects of the source structure change, the irradiation time error was evaluated and compared with the results of initial performance test in 2012.

For estimating the irradiation time error, Bonner sphere neutron spectrometer (model 42-5, Ludlum measurement, Inc.) was used. The detector was installed at a distance of 1500 mm from the neutron source. Total neutron counts were measured by increasing the setting time from 100 s to 500 s and the each set of measurements was repeated three times. The measurement time was added more than approximately 100 s to the setting time in the consideration of the source transit time. And then the simple linear regression analysis between total neutron count and the setting times was carried out. The counts when the setting time is equal to zero according to the linear regression formula could be evaluated as those occurred during the source transit time and the irradiation time error could be calculated using those counts and a certain neutron count rate.

Table 1. Results of the irradiation time error before/after changing the source holder

	Before changing the holder (2012)	After changing the holder (2019)	
Count (when t = 0,) 1)	68876	777.1	
Applied count rate [cps]	4604.49 2)	45.02 2)	33.89 3)
Timer error [s]	14.96	17.26±1.53	22.93±2.03
Difference [s]	-	+ 2.30	+ 7.96

1) Counts caused by the neutron fluence during the source transit time
2) Count rate from the total neutron count measurement including the scattered neutron
3) Count rate from the direct neutron count measurement

The results are shown in Table 1. When the total neutron count rate was applied considering the calibration environment of personal dosimeter, the irradiation time error was assessed to be (17.26 ± 1.53) s and the difference from the initial performance test result was approximately 2.30 s. In addition, the irradiation time error was assessed to be (22.93 ± 2.03) s when the direct neutron count rate was applied considering the basis of the reference personal dose equivalent calculation and the difference from the initial performance test result was approximately 7.96 s. However, those difference depending on the count rate could be minimized when the dosimeter exposed for at least 1000 s and it is found that the contribution of the neutron fluence during source transit time could be less than 1 % when the dosimeter exposed for approximately 2300 s.

Keywords*: Neutron dosimeter, Personal dose equivalent, Irradiation time error*

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PS2 (T2.1-0943)

Workplaces radiations doses in X-Ray Facilities in southern Benin

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Radiation exposure from diagnostic radiology is fast increasing and it has become a growing concern[1] Study of the workplace is therefore necessary to ensure optimal protection for workers, particularly in countries where dosimetric monitoring does not exist. The purpose of this study is to assess ambient radiation doses at workplaces in X-Ray facilities in southern Benin in order to ensure that the exposure levels are below regulatory limits.

A cross-sectional study was carried out from June to July 2019 in Benin. This study was reviewed and approved by the authorization of the Research Ethics Committee (CER-ISBA) of Cotonou. The consent of the Directors of health establishments were requested and obtained before the measurements were carried out. An AT1123 radiameter from APVL[2] was used to measure exposure levels.

The scattered X-ray doses were measured, using a phantom at patient position, behind the leaded screen at a point situated horizontally 50 cm behind the screen and vertically 50 cm above the examination table, considered as the usual level of chest of health professional (as illustrated on the figure below). The X-ray examination simulated was lumbar radiography with different incidences (face and profile). A total of 33 x-ray rooms had participated in this study. The cumulative dose had been estimated on the basis of the weekly workload representing 70% of the total workload of the radiology room.

Out of 33 dose equivalent rate measurements carried out behind the leaded screen (worker's location) 30 were above a reference level set at 7.5 µSv/h at international level. Only 3 measurements were below this level. When we consider the cumulative dose equivalents based on the workload per room expressed in milliAmpere.minute per week, we find that all the cumulative dose at the same measurement points are lower than 0.3 mSv per week. This was achieved despite the existence of maximum dose equivalent rates of up to 141 mSv/h. This is explained by the low weekly workloads.

The mean dose equivalent rate was 6.3 mSv/h (min: 7.0310^{-07} max:141) mSv/h higher than those obtain in Achuka's study[1] in Nigeria in 2019, Getrude Chinangwa's study[3] in Malawi en 2017.

The dose equivalent rate of the workplace exposure level exceeded the recommended reference level, which suggests that standard lead glass shielding is insufficient and dosimetric monitoring is required.

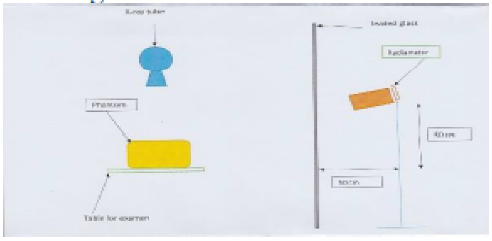

Fig. 1. Position of the radiameter during measurements.

Keywords: Assessment, Workplace, Radiation Dose

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PS2 (T2.1-0946)

A systematic review of radiation dose monitoring among radiology healthcare workers in Africa

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The past few decades have witnessed a wide substantial revolution across a wide array of diagnostic techniques.[1] Radiation exposure from diagnostic radiology is fast increasing and it has become a growing concern.[2]

Dosimetric monitoring is one of the means of prevention to limit health effects of ionizing radiation. The scarcity of publications on this topic in Africa, as shown by scientific literature, leaded us to undertake this study, which is to make inventory of the monitoring of external radiation exposure of healthcare workers on this continent.

Using standard terms, we conducted searches in PubMed/MEDLINE, Google Scholar and INIS databases. Two reviewers screened the retrieved publications based on predefined eligibility criteria to identify relevant studies, extract key information from each, rate the quality of evidence, and summarize data in a table.

A total of 18 potentially relevant articles were identified, with twelve articles that reported the overall mean annual dose and eight articles that reported collective doses. Studies included in this systematic review represent, across various countries in Africa, an inventory of the radiological protection of medical workers in Africa, with a focus on the monitoring of occupational radiation exposure. The size of study populations ranged from 81 to 5,152 healthcare workers. Many authors reported that the thermoluminescent dosimeter was the main monitoring equipment of healthcare workers. 41.67% of studies have a mean annual dose less than 1 mSv, despite the fact that workers are assigned to work with ionizing radiation. Annual mean doses were low in the medical group as compared with the industrial and research groups of workers. In some countries, occupational exposure decreased; in Ghana,[3] the collective dose has decreased by half for the same group of diagnostic radiology workers over a 9 years period.. Existing evidence on dosimetric monitoring in Africa aimed at reducing workers radiation dose is sparse.

Table. 1. Proportion of studies with a mean annual dose compared to the regulatory limit for the public.

Authors	Mean annual doses for medical sectors	
	< 1 mSv/an	> 1mSv/an
(Yakoro et al., 2010); (Tapsoba et al., 2012); (Sulieman et al., 2011); (Adjei et al., 2012); (Gordon et al., 2011)	5/12 (41.67%)	
(P. Farai and I. Obed, 2001); (Bayou, 1991); (Hasford et al., 2012); (Muhogora et al., 2013); (Korir et al., 2011); (Andriambololona et al., 2002); (Ogundare and Balogun, 2003)		7/12(58.33%)

Keywords: Monitoring Radiation, occupational exposures, Africa

ACKNOWLEDGMENTS

Special thanks to Team U1018 Gustave Roussy; INSERM and IAEA for financial of study.

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PS2 (T2.1-0949)

Patient dose reduction undergoing the most common CT examinations in Tunisia

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Objectives: The purpose of our study was to minimize patient dose undergoing the most common CT exams in Tunisia, with changing only one parameter was the target Standard Deviation (Target SD) of noise index.
Materials and methods: All CT examinations were performed by one facility, Hitachi optima 16 slices. Before collecting data, the $CTDI_{vol}$ displayed by the machine was verified with using PMMA phantoms for CT measurement (body and head phantom) and 10 cm calibrated ionization chamber. Four body regions were concerned by this study: Head, Chest-abdomen, chest-abdomen-pelvis and Abdomen-pelvis. The work was done in two periods: before and after optimization. The total number of adult patients considered was 187 with weight ranged between 50 and 90 kg. For each exam, the parameters of exposition were registered: kVp, Pitch, time rotation, effective mAs, collimation and target SD and iterative reconstruction was used. For patient dose the size-specific dose estimates (SSDE) was calculated from the conversion coefficients as a function of effective diameter published by AAPM [1] were used.
Results: There was no significant difference in $CTDI_{vol}$ between displayed values and measured (p=0.199). For each exam the number of patients was more than 20. The Average SSDE values, before and after modification of noise index, were 62 and 31 mGy for head; 18.7 and 12.2 mGy for chest-abdomen; 18.4 and 13.3 mGy for chest-abdomen-pelvis and 17.6 and 13.3 mGy for abdomen-pelvis; respectively. The dose reduction rate was ranged between 24 and 50 percent for all CT exams.
Conclusion: The results of this study showed that the Target SD of noise index is an important parameter witch used for patient dose reduction with maintaining image quality adequate for diagnosis.

***Keywords**: patient dose, Target standard deviation, $CTDI_{vol}$*

ACKNOWLEDGMENTS
I would like to thank IAEA for dosimetry equipment witch used in this study.

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PS2 (T2.1-0950)

Physicochemical properties of uranium aerosols at a nuclear fuel fabrication plant

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Background: Occupational inhalation exposure to uranium aerosols in nuclear fuel fabrication need to be monitored, e.g. by using the Human Respiratory Tract Model (HRTM) developed by the International Commission on Radiological Protection (ICRP) [1]. Default model parameters for particle deposition in the respiratory tract and lung clearance (e.g. absorption to blood) for various uranium compounds are available, but physicochemical properties can vary depending on material process history. The production of fuel for light-water reactors can be carried out in different ways, including wet-chemical conversion of low-enriched (up to 5 % ^{235}U) uranium hexafluoride via ammonium uranyl carbonate (AUC) to uranium dioxide. Physicochemical data on uranium aerosols at sites utilizing the AUC route of conversion are scarce in the open literature.

Method: Personal cascade impactors (Marple 298) were used to sample and size-fractionate (cut-points 0.5-21 µm) aerosols at a nuclear fuel fabrication plant utilizing the AUC route of conversion. Sampling was carried out in the breathing zone of operators (to obtain as representative sampling as possible), working at four of the major workshops at the site. Impactor substrates were analyzed using alpha spectrometry. Furthermore, aerosols were investigated using scanning electron microscopy with energy-dispersive X-ray spectrometry (SEM-EDX). Dissolution rate experiments of uranium aerosols in artificial lung fluid (Gamble's solution) are ongoing. Uranium aerosols collected at different impactor stages (i.e. different aerodynamic size) are evaluated to determine lung absorption parameters.

Results: Isotope activity ratios (^{234}U/^{238}U) at the different impactor stages frequently varied, indicating different enrichment levels and thus contributions from different uranium batches (enrichment levels vary between batches). This suggests multimodal size distributions, and a result a coarse and fine fraction was assumed in the modelling work. Approximately 80 % of the sampled activity was associated with the coarse fraction (AMAD 15-19 µm), i.e. significantly higher than the 5 µm ICRP default assumption. The AMAD of the fine fraction was evaluated to 2-7 µm, however associated with large uncertainties. [2] During SEM-EDX investigations, particle sizes >10 µm were frequently observed, confirming the presence of large particles [3]. Preliminary results from the dissolution rate experiments indicate that lung absorption parameters vary with aerodynamic particle size.

Conclusions: The present work investigates physicochemical properties of uranium aerosols formed at a nuclear fuel fabrication plant utilizing the AUC route of conversion. A bimodal approach was found suitable to describe the aerosol activity size distributions. Most of the activity was associated with the coarse fraction (AMAD 15-19 µm). Aerodynamic size fractionation of aerosols appears to be appropriate to increase representativeness of dissolution rate experiments in artificial lung fluid.

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PS2 (T2.1-0978)

Investigation of Radiation Fields and Radiation Work Tasks for Establishing a Dosimetry Program for the Lens of the Eye in Korean Nuclear Power Plants

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The necessity of monitoring the lens of the eye in occupationally-exposed radiation workers is increasing because of the 2007 Recommendation of the International Commission on Radiological Protection (ICRP) on lowering the dose limits of workers. Therefore, the investigation of radiation fields and radiation work tasks inducing the exposure to the lens is required to establish a dosimetry program for the lens of the eye in Korean nuclear power plants (NPPs). According to the literature, radiation workers who might have radiation exposure to the lens of the eye are classified into three categories: workers under uniform radiation fields, workers under non-uniform radiation fields with high equivalent dose for the lens of the eye, and workers with a high equivalent dose for the lens of the eye due to beta radiation of 700 keV or more but with a low effective dose.

For workers in uniform radiation fields, even if only a whole-body dosimeter worn on the chest, it is likely to perform an appropriate evaluation for the dose to the lens; it is, thus, not necessary to wear an eye dosimeter and any additional monitoring procedures. According to the references from domestic and foreign nuclear power plants, most radiation fields are uniform, and it is unusual for workers receiving relatively high radiation exposure to the lens of the eye only compared with the whole body.

Workers who work under non-uniform radiation fields and are exposed to high energy beta radiation can have exposure to the lens of the eye. In both the light and heavy water reactors, non-uniform radiation fields are mainly formed in steam generators. Thus, the work tasks involved in steam generator maintenance are designated as radiation works, which can cause radiation exposure to the lens.

The results of this review will be used as the reference for field experiments to evaluate the dose contribution to the eyes during the radiation works in Korean NPPs.

Keywords: *Lens of the Eye, Equivalent dose, ICPR 103*

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PS2 (T2.1-0987)

Study on the Mean Energy of Natural Radiation in Different Areas of the High Background Radiation Area of Yangjiang, China

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Abstract: Yangjiang is the farmous High Background Radiation Area (HBRA) in China. We focus the mean energy of external exposure of different types of houses in the HBRA of Yangjiang in this study. Proposed in publications, reports of ICRP and ICRU and Chinese national standards, The conversion coefficients between air kerma and the photon fluence and irradiation dose are different due to different photon energy. It is convenient to convert different physical quantities by mastered the mean energy. Four dose groups were selected as low, medium, high dose and control. For each dose group, a certain number of earthen houses, brick houses, and stone houses were selected for research. Field measurement of each selected locations using a portable, electrically cooled, HPGe gamma spectrometer. We analyzed the obtained spectra using the efficiency calibration method without radioactive source. The composition of all natural radionuclides in different dose groups and different types of houses were measured and compared. The share of each energy photon in various houses based on the probability of different energy photons of each radionuclide was calculated. Then calculate the mean photon energy of the corresponding place. Table 1 shows the evaluation results of the mean external radiation energy of different types of houses in the HBRA of Yangjiang. According to the average energy results, there was no significant difference in mean energy between different dose group, the mean energy of a brick house is slightly higher than the other two types. The possibility of this result comes from the material composition of the brick as a building material.

Table 1. Mean energy of natural radiation in different building material houses of HBRA, China

Dose grouping	House type	Average energy(KeV)
Control	earthen houses	768.7
	brick houses	783.4
	stone houses	782.1
Low dose group	earthen houses	766.8
	brick houses	781.2
	stone houses	759.3
Medium dose group	earthen houses	770.6
	brick houses	791.8
	stone houses	—
High dose group	earthen houses	758.4
	brick houses	758.3
	stone houses	750.9

Keywords: *Conversion factor, mean energy,* efficiency calibration method without radioactive source

PS2 (T2.1-1025)

Evaluation of Beam-Matching Accuracy for 8 MV Photon Beam between the Same Model Linear Accelerators

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This study aimed to assess of beam-matching accuracy for an 8 MV beam between the same model linear accelerators (linacs) commissioned over two years. Per vendor's customer acceptance procedure (CAP) criteria, the second linac installed at December 2018 was beam-matched with the first linac installed at July 2017. During the CAP of second linac, the beam matching criteria are based on depth ionization curves as well as in-line and cross-line profiles measured in the vendor-defined prescribed geometry. For commissioning data for beam-matched linacs, the percentage depth doses (PDDs), beam profiles, output factors, multi-leaf collimator (MLC) leaf transmission factors, and the dosimetric leaf gap (DLG) were compared. In addition, the accuracy of beam matching was verified at phantom and patient levels. At phantom level, the point doses specified in TG-53 and TG-119 were compared to evaluate the accuracy of beam modelling in TPS. At patient level, the dose volume histogram (DVH) parameters and the delivery accuracy are evaluated on VMAT plan for 40 patients who included 20 lung and 20 brain cases. Ionization depth curve and dose profiles obtained in CAP showed a good level for beam matching between both linacs. The variations in commissioning beam data, such as PDDs, beam profiles, output factors, TF, and DLG were all less than 1%. For the treatment plans of brain tumor and lung cancer, the average and maximum differences in evaluated DVH parameters for the PTV and the OARs were within ±0.30% and ±1.30%. Furthermore, all gamma passing rates for both beam-matched linacs were higher than 98% for the 2%/2 mm criteria and 99% for the 2%/3 mm criteria. The overall variations in the beam data, as well as tests at phantom and patient levels remains all within the tolerance (±1% difference) of clinical acceptability between both beam-matched linacs. Thus, we found an excellent dosimetric agreement to 8 MV beam characteristics for the same model linacs.

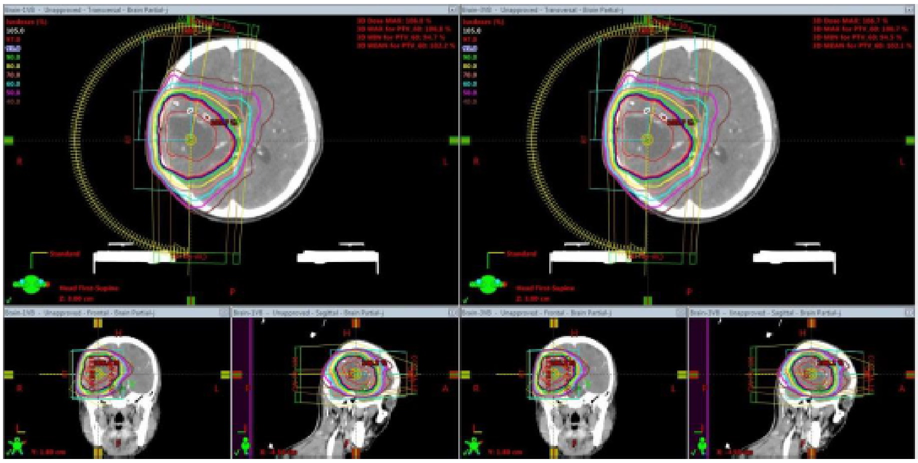

Fig. 1. Comparison of the dose distributions for the axial, sagittal, and coronal directions for brain VMAT plans using beam-matched linacs.

Keywords: Beam-matching, Commissioning, VMAT

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PS2 (T2.1-1034)

Organ Dose Assessment in Transarterial Chemoembolization Procedures Using Monte Carlo Simulation

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Transarterial chemoembolization (TACE) combines the local delivery of chemotherapy with embolization procedure to treat cancer, most often of the liver. It is a non-surgical and minimally invasive procedure performed in radiology, usually by an interventional radiologist. In TACE, anti-cancer drugs are injected directly into the blood vessel feeding a cancerous tumor. Synthetic material called an embolic agent is also placed inside the blood vessels that supply blood to the tumor, in effect trapping the chemotherapy in the tumor and blocking blood flow to the tumor.

The aim of this study was to assess organ and effective doses associated with TACE procedures using Monte Carlo computational program to simulate the procedure. Data on 15 patients (weight 82.5±14.8 kg; height 166.3±12.9 kgm^{-2}) who underwent transarterial chemoembolization procedures within an 11-month period, were extracted and simulated using PCXMC version 2.0.1.4. Input data for the simulation included patient weight and height, image height and width, projection angle, distance source-to-detector, distance source-to-patient, dose area product, X-ray tube potential, filtration and the anode angle of the X-ray tube.

From the simulated results presented in Table 1, the kidneys received the highest dose of 194.8 mGy (57.7 – 407.9 mGy). TACE is most beneficial to patients whose disease is predominately limited to the liver, whether the tumor began in the liver or metastasized to the liver from another organ. This accounts for the high doses of radiation to neighbouring organs like the kidney and adrenals. Organs such as the brain, extrathoracic airway, oral mucosa, prostate, salivary gland, testicles, thyroid and urinary bladder received relatively lesser dose levels. Estimated wholebody dose per patient of 33.3 mSv was found to be comparable to the effective dose based on ICRP 103 protocol.

Table 1. Simulated organ doses (mGy) from TACE procedures

		Min	Max	Mean ± δ	P90	Sum
Organ Dose (mGy)	Active bone marrow	5.69	31.70	17.26 ± 7.64	26.98	258.92
	Adrenals	42.89	227.18	132.22 ± 62.55	205.11	1983.31
	Breasts	0.83	4.76	2.29 ± 1.25	3.94	34.32
	Colon	2.89	13.07	6.85 ± 3.33	12.36	102.78
	Gall bladder	13.86	86.47	44.15 ± 21.21	67.46	662.18
	Heart	2.67	11.58	7.14 ± 3.36	67.46	107.08
	Kidneys	57.67	407.92	194.80 ± 106.11	343.08	2921.93
	Liver	24.71	327.38	122.50 ± 100.54	269.83	1837.49
	Lungs	5.51	38.29	18.98 ± 9.97	32.13	284.75
	Lymph nodes	4.96	20.65	12.76 ± 5.11	19.11	191.40
	Muscle	4.03	22.82	11.67 ± 5.67	19.80	175.12
	Oesophagus	2.98	20.99	10.24 ± 5.18	15.66	153.56
	Ovaries	0.71	11.76	2.55 ± 2.74	4.00	38.24
	Pancreas	5.16	43.02	20.87 ± 10.19	32.94	313.01
	Skeleton	8.91	38.92	23.40 ± 9.88	35.91	350.98
	Skin	4.35	28.35	13.50 ± 7.10	23.28	202.50
	Small intestine	3.39	15.77	8.81 ± 3.92	14.56	132.09
	Spleen	0.36	11.75	3.35 ± 3.03	6.84	50.27
	Stomach	0.78	9.53	4.06 ± 2.20	6.09	60.84
	Uterus	0.51	14.43	2.32 ± 3.42	2.79	34.76
Wholebody dose (mGy)		5.57	33.29	16.60 ± 8.49	28.62	249.04
Effective Dose ICRP103 (mSv)		5.27	30.39	15.81 ± 7.88	27.21	237.09

δ: standard deviation; P90: 90th percentile.

***Keywords**: Transarterial chemoembolization, PCXMC*

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PS2 (T2.1-1044)

Feasibility Study for Radiation Monitoring and Mapping System with SLAM-Based Mobile Robot

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In radiation-related facilities such as the radiation disposal repository, radioisotopes utilization facility or nuclear power reactor, radiation monitoring is an important issue for the radiation safety. Especially, for the monitoring a large region considering the exposure to the high level of radiation, it has a limitation to conduct the monitoring by human for the radiation safety and the inspection cost. An autonomous driving robot and IoT technique including the radiation monitoring system can be a solution to cover the large region monitoring without human effort. Also, a real-time radiation mapping is also utilized for establishing the working schedules of workers in the facilities. In this study, a feasibility on the radiation monitoring and mapping system using autonomous driving robot with Simultaneous Localization And Mapping (SLAM) technique [1] is tested.

For constructing autonomous driving and monitoring system, cartographer algorithm [2] was used for mapping and localization with two sub-systems: First sub-system is a local SLAM called as the frontend for constructing successive sub-map by scan matching; the second sub-system is global SLAM called as backend for the global optimization. With the SLAM technique, a prototype robot based on the turtlebot3 waffle pi was developed. For the radiation mapping and robot position localization, radiation detector, optical camera and RP-Lidar sensor were installed into turtlebot3 waffle pi as shown in Fig. 1. Also, OpenCR and LTE modem were used for transfering the radiation and local position information to the ground control system. With matching all information, the real-time radiation monitoring system was tested as the feasibility study. As a future work, a real-time radiation mapping system will be constructed after developing mapping algorithm with considering detector performance and optimizing the autonomous driving system.

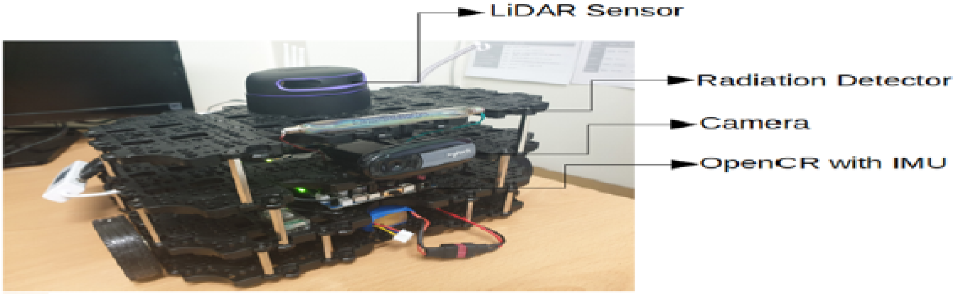

Fig. 1. Configuration of turtlebot3 waffle pi

Keywords: Unmanned Inspection System, Radiation Monitoring and Mapping, SLAM

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PS2 (T2.1-1064)

Feasibility of a Wrist-worn Wearable Device for Simultaneously Monitoring the Radiation Exposure and the Heart Rate of Radiation Workers

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Acquisition of the biomedical data, such as heart rate, oxygen saturation (SpO_2) and body temperature, as well as the measurement of radiation dose is helpful to promote the radiation safety and health of radiation workers. The aim of this research is to evaluate the feasibility of the wrist-worn wearable device for simultaneously monitoring the radiation exposure and the heart rate of radiation workers. This type of device consists of three modules including a radiation detector module, a pulse oximeter module and a microcontroller (MCU) module. Here, the radiation detector module employs the Geiger–Muller tube (G-M tube) to sensitively measure the exposure doses of radiation workers. In case of the pulse oximeter module, the high-sensitive photoplethysmography (PPG) sensor which operates in an ultra-low power environment is used to monitor the heart rates of the radiation workers. The MCU module contains a 32-bit microprocessor and a dual-mode Bluetooth controller that can communicate with a smartphone, a personal computer or another Bluetooth controller. The device is powered by a 3.7 V lithium-ion battery, which can be charged simultaneously from a USB power source. In order to evaluate the performance of the radiation detector module, Cs-137 isotopes with the activities of 1 μCi and 5 μCi were used as the radiation sources. To verify the normal operation of the pulse oximeter module, the heart rate was measured at the researchers' left wrists. The count rate was measured using the device as functions of a distances from the sources and radioactivities of the sources. As shown in Fig. 1, the measured count rate decreased according to the distance and was proportional to the radioactivity. The heart rate measured on the researchers' left wrists was not statistically different from that obtained from electrocardiography (ECG).

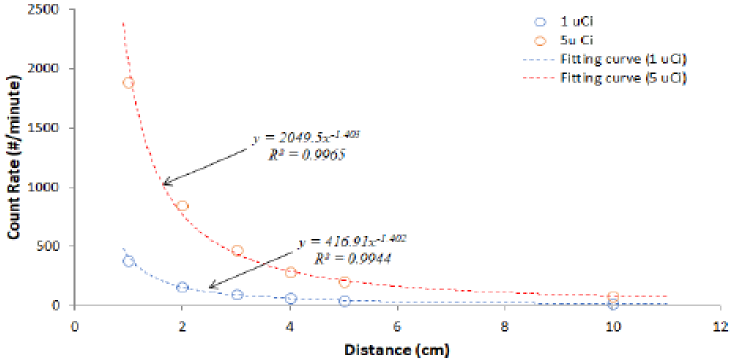

Fig. 1. Measurement of the count rate in various distances and radioactivities.

Keywords: *Wearable, Radiation, G-M tube, Heart rate, PPG*

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PS2 (T2.1-1079)

Establishment for Personal Dosimetry Performance Test by New ANSI Criteria in Korea

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The radiation exposure dose of radiation workers is the basic data for radiation protection and is used as a key basis for effective implementation of radiation safety management. In 1995, Korea introduced the performance inspection and quality assurance system for the reading of personal dosimeters for the first time in Korea. The regulations related to reading have been newly introduced in the nuclear safety law, and the enforcement rules of nuclear safety law and enforcement decree of nuclear safety law are in place to maintain the reliability of the reading performance test. However, this standard is based on the technical standard for performance evaluation of individual dosimeters in the United States, "Personnel Dosimetry Performance-Criteria for testing" (ANSI N13.11-1993). Since 1993, changes in the radiation environment in the United States have been revised in 2001 and 2009, and are now applying the latest standards revised in 2009(R2015). Since the introduction of dose reading technology in Korea, various changes have occurred in the radiation environment, and it is necessary to establish a new technical standard related to read performance reflecting such changed environment. Therefore, the purpose of this study is to develop a technical standard for the personal dosimetry performance test suitable for Korea using the latest standards of ANSI N13.11.

The research team has established a suitable technical standard for Korea by forming a council where experts from related fields participate in improving the domestic personal dosimeter performance inspection system.

The most fundamental basis for determining the performance test category is to make it easier about users to understand. The purpose of this study was to maintain the form of the performance test category used in Korea and to minimize confusion in information such as radiation fields.

In order to reduce the number of dosimeters required for the performance test and to prevent the difference in reading difficulty for each institution, the technical criteria of the individual dosimeter reading performance test will be established.

Keywords: Personnel dosimetry, Performance testing, ISO radiation fields

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PS2 (T2.1-1080)

Radiation Survey of Tc-99m Occupational Exposure in a Tertiary Hospital in the Philippines

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Exposure to low dose ionizing radiation has increasingly become more common over the past 25 years because of medical procedures. Nuclear medicine is where the patients, staff, and public can be exposed to low dose ionizing radiation. Single Photon Emission Computed Tomography (SPECT) imaging procedures consist of the patient receiving a dose of a radiopharmaceutical, making the patient a radioactive source. Thus, interactions with the patient results to added occupational exposure. This study aimed to determine the highest radiation exposure that can be received by the technologist interacting in close proximity with patients undergoing Tc-99m imaging studies in the SPECT room of the hospital and evaluate the doses received whether they are within the limits set by the International Commission on Radiation Protection (ICRP). Four sets of measurement were then taken when the Tc-99m source is: (1) inside the phantom and collimators, (2) inside the phantom and outside the collimators, (3) outside the phantom and inside the collimators, and lastly (4) outside the phantom and collimators. The dose rates were taken using a calibrated survey meter at various distances from the SPECT at 0.5 m intervals. At each location, 5 readings were taken at 5 second intervals, serving as the trials for this study. The annual effective dose was then computed for 5 mins of interaction with the patient and hospital's daily average of 10 SPECT imaging patients. This was then compared to the ICRP standard for occupational exposure, 20 mSv/yr. The phantom and collimators significantly attenuated the gamma radiation emitted by the source at α=0.01. The collimators absorbed most of the radiation given the significant decrease in the effective dose calculated between the inside and outside the collimators which is 9.15 and 9.94 mSv/yr for outside and inside the phantom, respectively. The highest dose that could be received by a technologist is directly adjacent from the patient couch, wherein 24.86 mSv/yr and 27.61 mSv/yr were the highest computed dose for the right and left side of the patient couch, respectively. Before and after the scan, the technologist is usually at the right side of the patient couch preparing the patient; such was simulated by having the Tc-99m inside the phantom and outside the collimator. The calculated effective dose is 23.11 mSv/yr, which is higher than the limit by 3.11 mSv/yr. However, by decreasing the interaction time to half, the resulting dose is 11.58 mSv/yr and is thus significantly lower than 20 mSv/yr at α=0.01. Also, the decrease in dose as distance from the source increased is explained by the inverse square law. So, at 1.5 m from the source, the highest annual effective dose that can be received by the technologist is 1.02 mSv/yr. In conclusion, decreasing the time spent in close proximity with the patient can significantly reduce the effective dose received by the staff. Moreover, increasing the distance of the technologist to the patient even by 0.5 m can effectively decrease the dose received.

Keywords: Occupational Exposure, Tc-99m, SPECT

ACKNOWLEDGMENTS

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PS2 (T2.1-1093)

Estimation of the skin exposure of the hands of workers handling selected radiopharmaceuticals using a finger dosimeter applying a correction factor

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Introduction: Over the past few years, new devices for the preparation or application of radiopharmaceuticals have been put into operation in order to reduce the exposure of hands compared to manual methods. Despite the fact that many workplaces are equipped with modern technology, there are situations when a worker gets contaminated or performs a non-standard working procedures, which could lead to the elevated exposure during such operations and thus, to contribute significantly to the higher annual exposure. Increased exposure may not be reflected in the results obtained from finger dosimeters, which are usually worn on the roots of the fingers, since the highest exposure is often located at the tips of the fingers. For this reason, it is necessary to introduce a correction factor defined as a ratio of the exposure at the site of the assumed maximum irradiation (most often at the tip of the index finger) to the exposure at the point where the finger dosimeter is worn (the index finger root).

Aim: The purpose of the paper is to overview the skin exposure related to the use of some new technologies developed over the last 10 years. At the same time, the evaluation of the local skin exposure and the correction factor associated with the manipulation involving ^{68}Ga-labelled positron radiopharmaceuticals is to be analysed and compared with the situation characterized by ^{18}F-labelled radiopharmaceuticals.

Material and methods: The selected nuclear medicine departments using ^{18}F and ^{68}Ga labelled radiopharmaceuticals participated in this pilot study. Thermoluminescent dosimeters placed at 10 locations on the hand were used to determine $H_p(0.07)$. The maximum exposure of each worker's skin was related to the activity being manipulated. Subsequently, the average values related to the position at the tip of the index finger were compared for selected professional groups (radiopharmaceutical labelling, radiopharmaceutical filling into syringes and radiopharmaceutical application) and a correction factor for monitored workplaces was determined.

Results: The paper presents a summary of the comparison of results from finger dosimeters in selected occupational groups in the period of 2009 - 2019 and also highlights new findings regarding the correction factor for selected radiopharmaceuticals over this decade. In addition, the pilot experiments with ^{68}Ga are discussed in terms of the average skin exposure in the position at the tip of the index finger of the workers. The paper also illustrates the specific cases of manipulation with ^{68}Ga-labelled radiopharmaceuticals, which resulted in elevated exposure of the skin.

Conclusion: A more detailed monitoring of the skin exposure at more locations on the workers' hand can reveal cases of abnormal work operations or significant local exposure of the skin, which is often found on the fingertips. Based on these findings, the appropriate measures can be taken to eliminate non-standard operations in a timely manner and thus minimize this exposure. If adequate correction factors are used, the exposure at the tip of the finger can be straightforwardly estimated from the results of the finger dosimeters worn at the root of the finger.

Keywords: Skin exposure, Correction factor, Nuclear medicine

ACKNOWLEDGMENTS
The paper was partially supported by project SGS18/00/OHK4/1T/17.

PS2 (T2.1-1099)

Investigation of a surface dose distribution on the CIRS human phantoms for KREDOS intercomparison

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When a non-destructive test (NDT) using a high activity isotope is conducted, radiation workers are in a potential accident that can lead to a significant exposure. Under the framework of Korea retrospective dosimetry (KREDOS) inter-laboratory comparison held in 2019, a thermoluminescence (TL) retrospective dosimetry of an accident scenario was investigated in the radiation testing (RT) room. To verify a distribution of surface dose on victims under the working condition, two CIRS ATOM phantoms (Adult male (model 701) and adult female (model 702)) were irradiated by a 32.99 Ci Ir-192 source in the RT room. The female phantom was irradiated for 2 hours at 30 cm from the source, and the male phantom was irradiated for 7 hours at 1 m from the source. Dose rates measured by electronic personal dosimeters (EPD) were 1.8 and 0.19 Sv/h on the front side of the female and male phantom, respectively. The dose rates on the back side of the phantoms were dropped to 0.2 Sv/h for the female and 0.028 Sv/h for the male due to a body shielding. In order to verify surface dose distribution, LiF:Mg,Cu,Si TL dosimeters (TLDs) were attached on the surface of each phantom. Table 1 shows measured doses according to the position. The values are Cs-137 equivalent air Kerma. Since TLD is one of the most commonly used reference dosimeters, the result can be used as a criteria for the harmonization of various other dosimetry techniques. Moreover, the results will be applicable as a reference frame of comparison between measurements and calculations carried out in the KREDOS inter-laboratory comparison.

Table 1. The measured dose using TLDs on the surface of the human phantoms [unit: Gy]

Location	Female						Male					
	Front			Back			Front			Back		
	Right	Middle	Left	Right	Middle	Left	Right	Middle	Left	Right	Middle	Left
Head	-	1.10	-	-	0.15	-	-	0.92	-	-	0.13	-
Shoulder	1.65	1.80	1.56	–	–	–	0.96	1.03	0.96	0.29	0.21	0.23
Chest(upper)	2.13	2.26	2.33	0.64	0.28	0.42	1.10	1.12	1.05	-	-	-
Chest(lower)	-	-	-	-	-	-	1.09	1.12	1.05	0.38	0.18	0.35
Waist	2.27	2.85	2.45	0.40	0.39	0.37	1.00	1.11	1.04	0.20	0.18	0.19
Hip(upper)	-	-	-	0.20	0.21	0.20	-	-	-	0.15	0.14	0.12
Hip(lower)	1.32	1.36	1.39	0.16	0.18	0.15	0.97	1.03	0.97	0.12	0.14	0.13

Keywords: Phantom surface dose, KREDOS, Irradiation experiment

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PS2 (T2.1-1102)

Exposure optimization caused by handling of radiopharmaceuticals

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Introduction and aim: Handling of radiopharmaceuticals during their preparation and application consists of a number of operations which, depending on the geometry and duration of irradiation, cause various exposure of the skin on the hands. The results of personal monitoring show that approximately 10% of workers at nuclear medicine departments in Slovakia and in the Czech Republic regularly receive an increase skin exposure which is close to the investigation level set by the respective regulatory authorities. The reasons for such elevated exposure are not usually discovered and identified. The presentation suggests a new way of identifying a probable cause associated with the specific procedures during the manipulation that may have induced excessive skin exposure.

Method: During the manipulation with radiopharmaceuticals at nuclear medicine departments, the hands of the personnel close to the sources handled are inhomogeneously exposed. The contact with radioactive sources include preparation and application of radiopharmaceuticals to the patients. The ORAMED project found a direct relationship between the total distribution of the personal dose equivalent $H_p(0.07)$ characterizing skin exposure when preparing or administering radiopharmaceuticals to patients using various individual approach or techniques. For a worker with excessive skin exposure, in case of overexposure the dose distribution is visibly distorted. By comparing the analysis of this distribution using the multiple linear regression (taken into account spatial distributions from individual operations) with a similar analysis of a worker normally exposed under the same radiopharmaceutical manipulation procedures, it is possible to identify a possible work action causing excessive exposure of the skin of a monitored worker.

Results and discussion: The dose distribution matrix of 12 selected radiopharmaceutical manipulation operations when administering the FDG radiopharmaceutical to a patient was determined by the measurement on a physical phantom in combination with Monte Carlo calculations using MCNP5 code and a voxel hand phantom. The gloves covered with 10 thermoluminiscent dosimeters were used to map the overall dose distribution on the skin of workers during the administration. The method of identifying the operations or procedures causing excessive exposure was used for the investigation of two overexposed workers engaged in administering FDG radiopharmaceuticals to the patient. As to the first investigated worker, the most serious procedures behind the excessive skin exposure was attributed to the non-reported administration of FDG to patients by unshielded syringe while in the second case, the overexposure was caused by the undetected surface contamination of the hand.

Conclusion: The proposed new method of the optimization of handling of radiopharmaceuticals allows for effective disclosures of the cases of reducing hand exposure in individual steps of handling radiopharmaceuticals related to their preparation or application to patients.

Keywords: Skin exposure, MCNP5, Nuclear medicine

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PS2 (T2.1-1105)

Calibrating a Neutron Survey Meter in Various Standard Fields

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In this work, the calibration of a neutron survey meter has been performed in several neutron standard fields (i.e., two neutron standard fields of bare ^{252}Cf and $^{241}Am\text{-}Be$ sources, and five simulated workplace fields of $^{241}Am\text{-}Be$ moderated sources) so that the calibration factors (CFs) of the neutron meter have been evaluated. In order to calibrate the neutron meter in the standard fields of bare sources, the total neutron ambient dose equivalent rates ($H^*(10)'_{tot}$) measured by neutron meter were analyzed to obtain the direct component ($H^*(10)'_{dir}$) using the reduced fitting method as the ISO 8529-2 recommendations. The CFs were then determined as the ratio between the conventional true value of the neutron ambient dose equivalent rate in a free field ($H^*(10)'_{FF}$) and the value of $H^*(10)'_{dir}$. Otherwise, the CFs of the neutron meter in the simulated workplace neutron fields following the ISO 12789 series recommendations is determined as the ratio between the values of $H^*(10)'_{tot}$ measured by a standard instrument (i.e., Bonner sphere spectrometer) and by the neutron meter to be calibrated. As results, the CFs were in the range of 0.88–1.00. The uncertainties of the CFs were determined of about 6.6–13.1% following the uncertainty propagation principle.

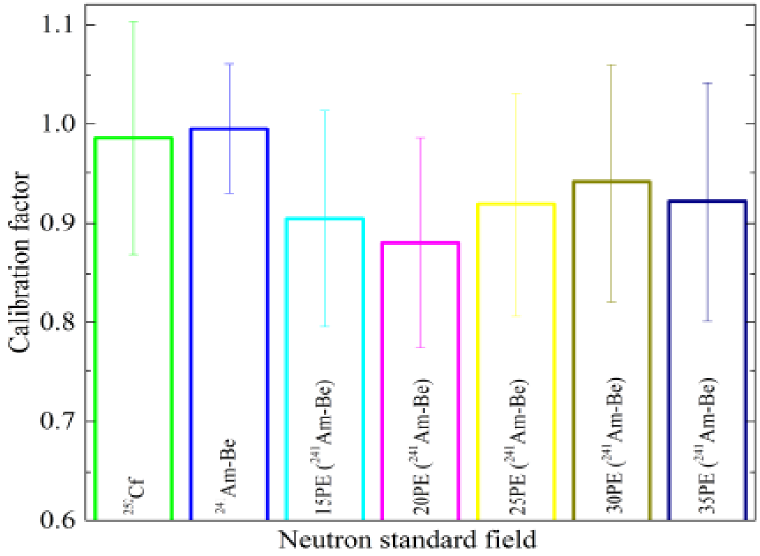

Fig. 1. Calibration factors and its uncertainties of the neutron meter in various neutron standard fields. 15PE(^{241}Am-Be) means the ^{241}Am-Be source moderated using a polyethylene sphere with a diameter of 15 cm.

Keywords: Neutron standard fields; neutron meter; calibration factors

ACKNOWLEDGMENTS
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PS2 (T2.1-1106)

EVALUATION OF THE EFFECT OF FILTERS ON RECONSTRUCTED IMAGE QUALITY FROM COME BEAM CT SYSTEM

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Abstract: 3D Filtered Back Projection (FBP) is a three-dimensional reconstruction algorithm usually used in Cone Beam Computed Tomography (CBCT) system. FBP is one of the most popular algorithms due to its simplicity. FBP can produce 3D reconstructed objects much quicker. It can also handle a more considerable amount of data while not requiring powerful computer hardware than other algorithms. The quality of a reconstructed image by the FBP algorithm strongly depends on spatial filters and denoise filters applied to projections. This paper will evaluate the reconstructed image quality of the CBCT system by using different denoise filters and spatial filters and then finding out the best filters for the CBCT system.

Keyword: *Cone-Beam CT, FBP, Filter, Reconstruction*

PS2 (T2.1-1111)

Fuzzy Analysis of Radiologic Doses Data for the Estimation of Diagnostic Reference Levels in Spain (Project DOPOES II)

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The project DOPOES II works **for estimating the diagnostic reference levels of radiologic doses in Spain** through the analysis of more than twenty millions of medical records which have been read from different medical and hospital data bases systems, HIS (Hospital Information System) and RIS (Radiological Information System) environments. The key data for calculate the reference levels is the dose but this information can be completed with lots of information about the patient using DICOM (Digital Imaging and Communication On Medicine) header such as age, sex and BMI (body mass index) for example, and other complementary data as type of medical process, modality and information about the dipositive (calibration parameters, manufacturer, model, …) in order to create a complex data warehouse which it must be optimized and analyzed using business intelligence tools to obtain useful medical information.

Traditional methods of analysis cannot assume the massive amounts of data distributed across different locations, the DOPES II works with information of more than 50 public and private hospitals from all Spanish territory, for these reason new techniques based in artificial intelligent must be used to exploit the information and create new knowledge to improve the patient safety.

The information collected in DOPOES II is analyzed and tested in order to guarantee the quality of data. In the initial phases different techniques have been tested for imputation and clear of information. Finally, a fuzzy imputation system with majority ordered weighted averaging aggregation operators (MA-OWA) combined with business intelligence tools are used to test and validate the data base and build advanced queries and models that visualize the data.

Also, other application of artificial intelligence emerges from the project when the whole available information is studied. To identify mistakes in the different levels of information we propose neural networks. Using a multilayer perceptron model for evaluating the system is able to detect unusual doses values having as control variables the parameters with influence in the amount of dose: age, procedure, modality, sex and BMI. The doses data combined with real DICOM information can provide a huge set of training examples with their corresponding correct responses which can be used for supervised learning.

Keywords: *Radiologic doses, Artificial Intelligence, Fuzzy systems. T4. Practical Implementation: Medical Sector*

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Poster Session T2

PS2 (T2.1-1112)

Diagnostic reference levels for mammography in Spain. Preliminary data from Project DOPOES II

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The project DOPOES II is developed through a specific collaboration agreement between the Nuclear Safety Council (CSN) and the University of Malaga (UMA), whose objective is to estimate dose reference levels (DRLs) in medical radiodiagnostic procedures to patients in Spain, as well as its contribution to the doses received by the population. According to the results of the DOPOES I project, mammography is the second most frequent procedure in medical radiological procedures in Spain. In this study, dosimetry data from 31 health centers in 17 Autonomous Communities has been analyzed.

The results have been obtained for 2D (DR) and 3D digital mammography equipment (tomosynthesis) equipped with RIS and / or PACS management system, obtaining from them the following data: characteristics of the mammography used, age, exam date, mean glandular breast dose (MGD), Entrance Surface Dose (ESD) and compressed breast thickness (CBT). Examinations performed on both breasts with Craniocaudal view (CC) and Mediolateral oblique view (MLO) have been included. Also, the difference in MGD in tomosynthesis equipment versus 2D equipment has been studied.

The study covers more than 150,000 patients underwent mammography studies during a 2-year period (from 2016 to 2017). The average age has been 55 years (min 18 years; max 98 years). For the total of the sample studied, the MGD has presented significantly higher values for the OML than for the CC. No differences were found between right or left breast. The average compression thickness has been 55 ± 5 mm. A proportional relationship between greater compression and lower dose has been found. No correlation was found between age and MGD.

The age range with the highest proportion of examinations is coincident with the age groups to Spain´s breast cancer screening programs. The total MGD of the sample stands at approximate values to those obtained with previous studies. The MGD has higher values for the MLO projection, likewise, as other studies have referred occurs when the MGD and the compression thickness are compared.

The DRLs have been obtained from the median of the distribution of MGD in the 75th and 95th percentile, since there are authors who use both percentiles to establish these values, 75th for international and local reference levels and 95th for local levels. Even in other publications, the amount used has been the MGD exclusively for OML projections. Also the reference doses from a compression thickness (CBT) of 53 mm as established by a European guide. The DRLs obtained are compared and discussed with those published in our European environment.

Keywords: Diagnostic reference levels, mammography

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PS2 (T2.1-1120)

Initial experience in Quality Control in Radiodiagnosis, dose evaluation in Guatemala

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In Guatemala there is the "Law for the control, use and application of radioisotopes and Ionizing Radiations" on the basis of which the "Regulation of safety and radiological protection of the law", it was created within the Governmental Agreement 55-2001. Within this regulation created in 2001, quality control tests are requested for ionizing radiation generating equipment. It was until 2016, when FIXCA S.A. conducted the first evaluations of ionizing radiation generating equipment used in diagnostic radiology, which complies with said requirement to obtain the respective permits. These tests were performed at the request of a Hospital Network that wanted to provide better quality of service to their patients, verifying that their radiodiagnosis equipment complied with the tolerances recommended internationally.

Since then, FIXCA has provided the quality control service of ionizing radiation generating equipment, helping the providers of this service to make the respective corrections as the case may be. Or provide a report that indicates the values and ranges of acceptance of the tests performed.

This document shows the correlation carried out between the Raysafe Xi solid state detector and the TLDs 100 so that we can obtain the dose only with the detector, without the need to perform the TLD tests on each device. First, a correlation of the dose reading of the TLDs was carried out, assigning a factor to that reading for the delivered dose value. Subsequently, a series of irradiations with different exposure times showed in the figure 1 was carried out, in which we found a perfect correlation between the TLD readings and the RF Xi readings. Within the tests carried out we also show the kV values obtained in the different RX equipment. The values obtained in the equipment are also reported evaluating their values from 50kV to 110kV. And in correspondence of the calibration values, the input dose value is shown for the same equipment.

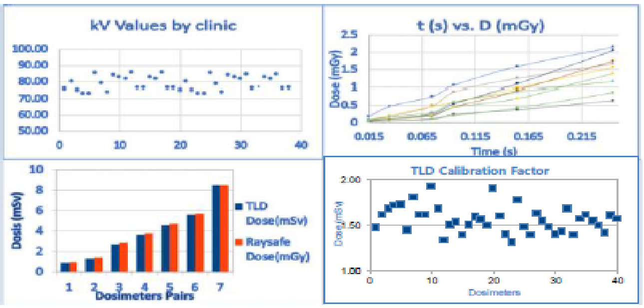

Fig. 1. The logo of the IRPA15

Keywords: *Radiodiagnostic dose, xRays, dose*

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PS2 (T2.1-1131)

Latest results and new directions in the field of individual monitoring of ionising radiation – summary of the IM2020 conference

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The International Conference on Individual Monitoring (IM) is considered to be the main conference dealing with the field of individual monitoring of ionising radiation. The International Conference Series is initiated by EURADOS and organised every 5 years for the purpose of facilitating the knowledge sharing, providing opportunity to exchange experiences and propagate innovations and new developments in individual monitoring methods in the fields of medicine, research and industry. As part of the Conference Series, Helsinki (2000), Vienna (2005), Athens (2010) and Bruges (2015) hosted the previous IM Conferences. Latest event of this prestigious Conference series is held from 19 to 24 April 2020, in Budapest, Hungary. The IM2020 conference is jointly organised by EURADOS and the Centre for Energy Research.

The Conference covers all aspects of individual monitoring of ionising radiation with topics including:

- New developments in external dosimetry
- New developments in internal dosimetry
- Individual monitoring in medicine, research and industry
- Computational methods in individual monitoring
- Dose assessment for exposures in workplaces and homes
- Monitoring and dose assessment in emergency exposure situations
- Exposure to radon and its progeny
- Air- and spacecraft crew dosimetry
- Individual monitoring services, quality assurance
- Type testing, intercomparisons and results
- International and European standards and recommendations
- Education, training and networks on individual monitoring

Besides the key lectures and oral presentations poster sessions, technical exhibition, roundtable discussion and refresher courses make the program even more interesting and colorful. The expected number of participants is around 300, including professionals and young researchers from Europe and beyond.
Most important developements in the field of individual dosimetry and highlights of the IM2020 conference will be summarized in the presentation.

***Keywords**: personal dosimetry, internal and external dosimetry, new developments*

ACKNOWLEDGMENTS

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PS2 (T2.1-1132)

New Korean Physical Phantom Model for Radiation Protection and Occupational Dose Reconstruction

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Computational phantoms (models) are frequently used to calculate the absorbed doses to organs for the estimation of effective doses to workers in specific irradiation geometries in the working places. However, in some cases like nuclear power plants, the radiation source distribution is difficult to be characterized. So, in this study, physical phantom models for organ dose measurement were designed based on the new ICRP recommendations [1] and average body size of Korean workers.

The range of body size of an adult male and female was determined by the height, weight, and chest-width of aged 20-60 year group from the nation-wide body size survey (National Institute of Standards and Technology, Size Korea 2015) results. Computed Tomography images of 200 male and female patients whose body size corresponds to this range were obtained. One image set of each male and female was finally selected as a reference model, which was the nearest to the average value. For tissue substitutes, seven types of equivalent materials were adopted as soft tissue, bone and lungs. Compared to previous model [2], the HU (Hounsfield unit) values as well as the densities of new tissue equivalent materials were tested and applied.

A new Korean physical phantom model was successfully established for male and female adult worker. This study presented the 3-D size information for all critical organs for effective dose estimation. Further, the method used to determine the appropriate substitutes is based on physical density, weight, homogeneity of materials, and the interaction between radiation and materials during CT scanning. The new phantom will be used to estimate the organ dose distribution to workers for various radiation works for occupational dose reconstruction and radiation protection purposes.

Keywords: *phantom, dose reconstruction, occupational dose, radiation protection*

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PS2 (T2.1-1142)

Establish of the Reference X-ray Irradiation System for Personal Dosimetry Performance Test by New ANSI Criteria

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Personal dosimeters are the instruments used by radiation workers for monitoring and managing radiation dose. Photon energy responses can vary depending on the kind of dosimeters and survey meters used, and for their calibration, a reference radiation field is required. Korea Atomic Energy Research Institute (KAERI) has been designated a national radiation calibration and testing laboratory by the Korean government and has established reference radiation fields and a secondary standard dosimetry laboratory in accordance with Korean Industrial Standards (KS) and International Organization for Standardization (ISO) standards [1,2].

In the reference radiation fields of KAERI, all types of radiation such as gamma, beta, X-rays and neutrons can be used. In particular, the reference X-ray field is designed to produce and use almost 40 different types of beams including beam qualities offered by National Institute of Standards and Technology (NIST) in ANSI N13.11-2009 and reference beam qualities by ISO-4037 with the use of an X-ray generator and rotation-type additional filtration system.

In this study, 7 beam qualities of the Wide-spectrum (WS) series and 5 of the High air-kerma (HK) series were assessed, and the quantity of the reference X-ray field was investigated compared to the international standards of beam qualities.

Keywords: *Performance testing, ISO radiation fields, reference X-ray fields, ISO reference beam, Half-value layer measurements*

ACKNOWLEDGMENTS

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PS2 (T2.1-1167)

Urine Background Level Survey for Naturally Occurring Radionuclides of Normal People

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Internal dosimetry of victims is effective techniques used as critical information for medical treatments pertaining to radiation emergency [1]. In vitro bioassay and analytical methods using excreta samples are usually performed by conducting the radiobioassays. In case of artificial radionuclides (plutonium and americium), the background radioactivity level in the urine samples collected from normal people is hardly measurable if there is no exposure to radiation, while the intake of naturally occurring radionuclides can occur through air and food consumption. The background level in these natural contaminants should be corrected by measuring the collected urine sample, and this has also been suggested in previous studies [2].

To estimate the background level of naturally occurring radionuclides, urine samples collected from about one hundred Korean people were analyzed and the measurement results were evaluated in view of Korean's lifestyle. The target radionuclides for background survey are ^{210}Po, ^{234}U and ^{238}U, and ^{230}Th. The overall analysis procedure for the radionuclides in urine samples was established and verified using reference materials. The amount of sample and counting time were estimated following the criteria of minimum detectable activity (MDA) and dose assessment recommended by other studies. The participants were from local areas where there was no radiation work. The local areas were selected to show their characteristics according to the inland areas, coastal areas, and urban areas. In addition, participants from the area near the nuclear power plant were selected and compared with the analysis results of control groups.

The background levels were evaluated with environmental radiation level to check the trend between both results. Also, the results in this study were reviewed with those obtained in other studies. The relation between radioactivity survey results of food and background results of urine samples was discussed according to food consumption. The local area with relatively high radon activity in groundwater was selected. Generally, background level of urine samples shows a little high result according to the environmental values. The inhabitants participated in this study and the measurement results of urine background level and radon level in groundwater were compared to estimate the relation. The survey for participants was performed to investigate the effectiveness of lifestyle. Creatinine concentration of each urine sample was measured to normalize the background radioactivity of urine samples by each radionuclide.

Keywords: *Background survey, polonium, uranium, thorium*

ACKNOWLEDGMENTS

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PS2 (T2.1-1174)

Radiation dose distribution of the surgeon and medical staff on orthopedic Balloon Kyphoplasty in Japan

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Balloon Kyphoplasty (BKP), a new treatment for vertebral compression fractures, requires relatively short surgery time, but is performed under continuous fluoroscopy. Radiation exposure of Orthopedic surgeon, chiefly, fingers that cannot be avoided to be exposed them in fluoroscopy are expected to be at high doses. However, few reports are available on evaluation of BKP radiation exposure in Japan. To investigate the radiation dose distribution of BKP, we conducted a descriptive research among orthopedic surgeon and medical staff. Table 1 shows the annual dose estimates of BKP surgery by orthopedist. The orthopedic surgeon worn his own spectacles (not protective goggles for radiation), disposable protective gloves (0.022 mm Pb eq. or less), and a protective apron for the trunk.

Table 1. Estimated cumulative radiation dose to an orthopedic surgeon on BKP operations in FY2018

Attached portion of Dosimeter	Left ring finger (mSv)	Lens of eye on the left side (mSv)	Neck on the left side (mSv)	Chest on the left side (mSv)
Type of dosimeter (Chiyoda Technol Co., Ltd)	RPL (JK type)	TLD (DOSIRIS)	RPL (FX type)	
dose equivalent	Hp(7)	Hp(3)	Hp(10)	Hp(10)
23 cases[a]	50.37	8.27	3.91	1.15
28 vertebral bodies[b]	40.88	6.72	3.08	0.84

a. Estimated based on the number of cases
b. Estimated based on the number of vertebral bodies.

Keywords: *lens of eyes dosimetry, fingers dosimetry, Balloon Kyphoplasty*

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PS2 (T2.1-1183)

Research on High Sensitive Detection Techonolgy of Trace Nuclides

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Terrorism is one of the major threats to public security in the world and also is a serious challenge to world peace and development.In order to effectively prevent the explosion attack of terrorist, all countries in the world attach great importance to the safety control of explosives, especially the flow control and safety inspection of explosives. Tracer technique is an important solution to solve the problem ,western developed countries have proposed dozens of tracer solutions and invested heavily in research and development,there are many methods for the traceability and security detection technology of explosives , but most of them are in the stage of research.Therefore, researchers are trying to find an effective traceability technique to achieve control and traceability of explosive items.

In order to effectively solve the problem of pre-explosion detection of explosives, a trace amount of radionuclide is added to the tracer marker ,targeted to the rapid detection demand of radioactive trace marker in explosives,a high-sensitivity detection device is develped for the detection of γ ray emitting from explosives and is used for rapid and accurate detection of uncontrolled explosives before explosion. As a part of the intelligent control platform for uncontrolled dangerous goods, the device is applied in the field of dangerous goods control, realizes the rapid detection of uncontrolled explosives, and plays a role in preventing and deterring terrorists in social security.

The high-sensitivity detection device of trace nuclides is mainly composed of detector unit, electronic unit, data acquisition and processing unit, object transfer unit and alarm unit. The block diagram of device design is shown in Figure 1. The device adopts a channel type structure, four large-volume plastic scintillators are enclosed into a ring detection channel, and each plastic scintillator is combined with a photomultiplier to form a detection unit of device. Due to the trace radionuclides in the tracer marks, the measurement of trace radioactive materials is not only affected by the performance of the system instrument itself, but also by the surrounding environment of the workplace, Compton scattering and cosmic rays. In order to reduce the influence of the natural radiation background on the detection sensitivity of the device, a special shielding structure is designed around the ring structure detector to shield the external interference. Experiment shows that the shielding structure can effectively shield the natural radiation and the detection efficiency of device is imporved.

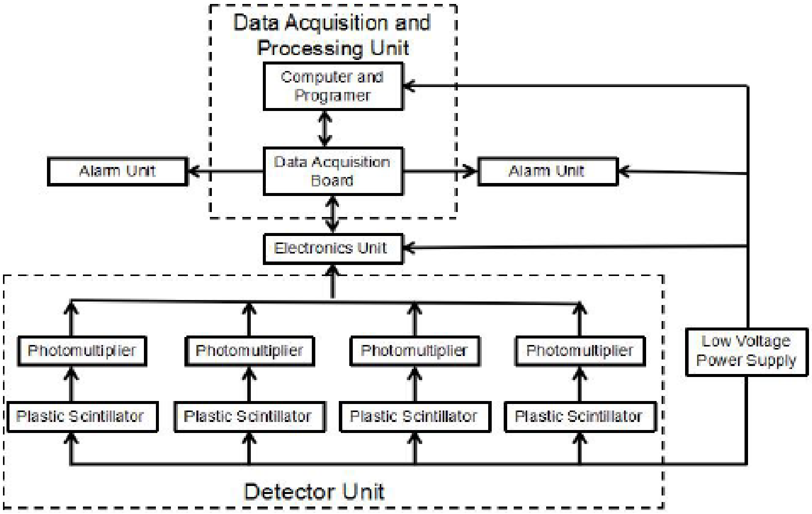

Fig. 1. Principle diagram of high sensitive detection system

PS2 (T2.1-1193)

Monte-Carlo simulation of clustered DNA double-strand breaks induced by alpha particles

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Radiation causes damage in DNA structure via energy deposition while passing through a cell nucleus. Low-LET radiation causes homogeneous DNA lesions, but high-LET ions often induce multiple double-strand breaks (DSBs) in proximity due to highly localized energy depositions along the track. Hardly repairable clustered DSBs apparently may lead to cell's mutation or death. In addition, such clustered DSBs can be perceived as a single DSB due to limitations in conventional measurement techniques.

In this paper, we defined a spherical nucleus volume in which the spatial distribution of DSBs induced by alpha particles were estimated considering particle energy and absorbed dose to the nucleus. The distributions of DNA damage were based on the track structure of alpha particles simulated using the Geant4-DNA. The probability density function of DSB separation and the DSBs falling within a clustering radius were assessed. The value of this study lies in describing the quality of DNA damage induced by alpha particles, and the potential applicability of our approach to examine the experimental measurement of DNA damage.

Keywords*: Alpha particle, Nano-scale Track structure, Clustered DSBs*

PS2 (T2.1-1198)

Estimation of Radiation Dose for the Korean Population by Dental Radiography

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Dental radiography is an important tool in modern medicine. Dental radiography is used for the diagnosis of cavities, implants, and orthodontics. The usage of dental radiography by the public continues to increase. This also increases the radiation dose for the public from dental radiography. Therefore, it is necessary to manage the radiation dose of Korean population by dental radiography. The objective of this study was to estimate the collective dose and effective dose per capita of Korean population by dental radiography. Dental radiography can be classified into intraoral radiography, panorama, and cephalo examination according to examination method. Collective dose and effective dose per capita for dental radiography were estimated based on usage and effective dose data of domestic dental radiography. The usage data of dental radiography in Korea was collected from the Health Insurance Review and Assessment Service (HIRA). The effective dose for intraoral radiography, panorama, and cephalo was used the value presented by NRPB-W4 report of the UK, the Ministry of Education, Science and Technology (MEST), and the Ministry of Food and Drug Safety (MFDS), respectively [1, 2, 3]. The total usage of dental radiography was about 30.8 million in 2017. The usage of intraoral radiography was the highest with about 19.6 million. The total collective dose for dental radiography was about 300 man·Sv. The collective doses for the examinations were 196 man·Sv (panorama), 98 man·Sv (intraoral radiography), and 0.096 man·Sv (cephalo). The total annual effective dose per capita was about 5.7 µSv. The annual effective doses per capita were generally high for panorama (3.8 µSv), intraoral radiography (1.9 µSv), and cephalo (0.002 µSv). This result can contribute to the management and optimization of radiation dose for dental radiography.

Table 1. Usage and radiation dose for the Korean population by dental radiography

Examination	Usage	Collective dose (man·Sv)	Annual effective dose per capita (µSv)
Intraoral radiography	19,605,427	98	1.9
Panorama	11,121,396	196	3.8
Cephalo	24,025	0.096	0.002
Total	30,750,848	294	5.7

Keywords: Dental radiography, Dose assessment, Medical radiation

ACKNOWLEDGMENTS
This work was supported through the KoFONS using the financial resource granted by NSSC. (No. 1803013)

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PS2 (T2.1-1208)

Determination of efficiency by mapping method using CT Scanner of the HPGe detector via Monte Carlo Simulation

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To get detection efficiency by the distance around detector, the full peak efficiency was used, and grid mapping method was used to investigate axisymmetric feature. It is measured in longitudinal direction of the detector up to ±5cm, while 0.5~5.5cm in lateral direction. The measured values were compared with those of computer simulation. The computer simulation is carried out both detector inner structure using computer tomography and manufacturer provided inner structure. Both values were same in statistical uncertainty level, and it is compared with that of manufacturer's inner structure.

The efficiency of cylinder beaker, Marinelli beaker and mapping method efficiency were compared in the middle of ^{134}Cs 604 keV detector. In the case of cylinder beaker's efficiency was 0.0412± 0.002, while the one of mapping was 0.040± 0.001, which conformed within the uncertainty level. The Marinelli beaker' efficiency was 0.0242±0.002, and the one of mapping was 0.0248±0.0013. It was found that if it was measured around each point of detector, it was possible to have accurate measurement on radioactive sources.

Key words: Mapping,,Monte Carlo simulation, Computed Tomography, HPGe

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PS2 (T2.1-1210)

Determination of efficiency by mapping method using CT Scanner of the HPGe detector via Monte Carlo Simulation

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Key words: Mapping,,Monte Carlo simulation, Computed Tomography, HPGe

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PS2 (T2.1-1214)

Reporting absorbed dose rate in air from Southern part of Ibaraki prefecture related to Fukushima Daiichi Nuclear Power Plant accident

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Fukushima Daiichi Nuclear power plant (F1-NPP) accident that occurred in March 2011which dramatically changed the distribution of environmental radioactivity in eastern Japan. Due to this accident enormous amount of radio-caesium released into the environment were estimated as 6-20 pBq [1]. After 10 years of F1-NPP accident, radio-caesium (^{134}Cs ($T_{1/2}$ = 2.06 y) + ^{137}Cs ($T_{1/2}$ = 30.17 y)) remains major concern from a radiological safety perspective. Ibaraki prefecture located ~ 180 km southeast of the F1-NPP have been selected to study the dose rate distribution in air. Therefore, the measurement of absorbed dose rate in ambient air were carried out using vehicle mounted 3" x 3" NaI(Tl) scintillation spectrometer (Car-borne survey) in August 2020. The absorbed dose rates in air ranged from 17.72 to 92.63 nGy h^{-1} with an average 41.78 ± 6.57 nGy h^{-1}. The estimated contribution of artificial dose rate in the air were vary from 0.02 to 34.96%. The annual external effective dose ranges from 0.1 to 0.56 mSv. The detailed information of the radiation survey will be discussed during the presentation.

***Keywords**: Fukushima Daiichi Nuclear Power Plant accident, Absorbed dose rate in air, Radiocesium, Car-borne survey*

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PS2 (T2.1-1219)

Bio-compatible internal shield for electron beam treatment of superficial lesions

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The purpose of this study is to evaluate dosimetric characteristics against an internal lead shielding covered by bio-compatible silicone rubber to achieve optimal electron treatment of skin and subcutaneous tumour lesions for head and neck. A novel bio-compatible and flexible silicone rubber, Ecoflex was used to cover the lead shielding as a backscatter electron absorber for electron beam radiation therapy (EBRT). The composition of Ecoflex was analyzed using X-ray fluorescence (XRF). The clinical 6 and 9 MeV electron beam from a Varian iX Silhouette LINAC was modeled in Monte Carlo (MC) simulation, performed with the MCNP6.2 code. The MC simulation was conducted to extract the phase space of the beam energy spectra for each incident electron energy and investigated the optimal thickness of Ecoflex to achieve the clinically applicable electron backscatter ratio [1]. The dose delivered to the upstream surface of Ecoflex and downstream were measured by a metal–oxide–semiconductor field-effect transistor (MOSFET) on the LINAC machine. The measured doses were compared with those calculated from MC simulations to verify the results. The MC simulation was validated with percent depth dose curves to the commissioning data. The comparison differences were under ±1% between the simulation and experimental results. Total optimal shielding thicknesses were 11 mm (7 mm upstream Ecoflex, 2 mm lead, and 2 mm downstream Ecoflex) and 15 mm (10 mm upstream Ecoflex, 3 mm lead, and 2 mm downstream Ecoflex) for each 6 and 9 MeV electron beam, respectively. Less than 10% of the dose enhancement at the upstream were measured using these shielding designs with over 95% reduction of transmission percent dose at the downstream. This study evaluated the feasibility of the internal lead shielding covered by Ecoflex on EBRT in cases of superficial tumour lesions such as lips. This result could be utilized effectively in the clinic.

Keywords: Internal shield, Ecoflex, electron backscattering, Monte Carlo simulation, MOSFET

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PS2 (T2.1-1222)

Sensitivity Enhancement of PRESAGE™ by Luminescence Lights Emitted from Inorganic Phosphor

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This study aims to improve the sensitivity of PRESAGE™ dosimeter. As the initiator molecules are activated by scintillation lights emitted from a phosphor, the amount of the excited molecules of initiators (i.e., tetrabromide carbon, CBr_4) is increased. When the PRESAGE™ dosimeter mixed with barium fluoride (BaF_2) is irradiated by MV photon beams, the fluorescent lights emitted from BaF_2 activate initiators more than the PRESAGE™ without BaF_2 do. Activated initiators are supposed to be radical molecules which could react with the hydrogen in the center of three aromatic ring [1]. Malachite green is used such as a reporter of dose absorbance by changing its color from colorless to green. The luminescence lights can be emitted from BaF_2 when BaF_2 exists in form of the lattice structure [2]. Solubility of BaF_2 was evaluated by two different approaches; 1) eriochrome black T (EBT) titration and 2) molecular dynamics simulation. In the case of EBT titration, as the ionized barium ions (Ba^{2+}) combine with EBT, the absorbance of EBT increased at the wavelength of 500 nm corresponding to its resonance frequency of the conjugated delocalized electrons. By using an UV/Vis spectrophotometer, we observed that the saturated concentration of the BaF_2 in PRESAGE™ solution was 0.04% by weight. In the other approach, Gromacs was used to simulate to calculate the potential of mean force based on Boltzmann factor. Then, an association constant was calculated by Kirkwood-Buff integrals method. The weight of BaF_2 retained as a solid lattice phosphor in Presage solution was finally decided. To verify that the enhancement effect is due to the scintillation of BaF_2, barium sulfate ($BaSO_4$) was added to PRESAGE™ and the sensitivity between PRESAGE™ with BaF_2 and PRESAGE™ with $BaSO_4$ was compared. The molecular concentration of Ba in PRESAGE™ with $BaSO_4$ was the same as the Ba in PRESAGE™ with BaF_2. The net optical density of the irradiated PRESAGE™ with $BaSO_4$ and PRESAGE™ with BaF_2 was evaluated. We demonstrated that BaF_2 could enhance the sensitivity of PRESAGE™ by its scintillating mechanism.

Keywords: Phosphor, PRESAGE™, sensitivity, scintillation

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PS2 (T2.1-1224)

Monte Carlo simulation of gold coated contact lens-type ocular *in vivo* dosimeter

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This study reports a sensitivity enhancement of gold-coated contact lens-type ocular *in vivo* dosimeter (*CLOD*) for low-energy X-rays. Monte Carlo (MC) simulations were performed to assess the dose enhanced by the gold layer on the *CLOD*. The human eye and *CLOD* were modeled in MCNP6.2 [1, 2], and X-ray sources were defined as 80, 120, and 140 kVp. The gold layer attached to a CLOD was ranged from 100 nm to 10 µm. The thickness of the active layer was 140 µm. The dose ratio between the active layer of a gold-coated *CLOD* and a *CLOD* without the gold layer, i.e., the dose enhancement factor (DEF), was calculated for three different X-ray sources. In addition, the differences in dose (DD) between the active layer of the *CLOD* and the lens with respect to the thickness of the gold layer were calculated. The DEFs of 100-nm to 5-µm thick gold layers increased from 1.7 to 5.4 for 120 kVp when the thickness of the active layer was 140 µm. The DEFs of a 10-µm thick gold-coated *CLOD* for three X-ray sources were lower than those of the 5-µm thick gold-coated *CLOD* owing to the self-absorption of low-energy secondary electrons. The DD was within 0.5% when the gold-layer thickness was less than 1 µm. For 5-µm thick gold-coated *CLOD*, the DD ranged from 1.8% to 2.6%. The MC results presented a higher sensitivity of gold-coated *CLOD* (approximately five times higher than that of *CLOD* without the gold layer). Gold-coated *CLOD* can be applied to evaluate the low dose delivered to the patient during CT imaging in the clinic.

***Keywords**: contact lens dosimeter, lens dose, Monte Carlo*

ACKNOWLEDGMENTS

This work was supported by National Research Foundation of Korea (NRF-2019M2A2B4095126).

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PS2 (T2.1-1240)

Radiation Exposure and Risks Estimation during Brain CT Scan in Children in Morocco

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Objective: Children with cerebral pathologies are frequently exposed to ionizing radiation during the diagnostic CT scans requested. Although radiation exposure is potentially carcinogenic, data on its effects are limited. This study aimed to assess the effective dose received during pediatric brain CT scan to estimate the risk of cancer and the heredity risk of this procedure.

Material and Methods: A total of 300 patients were examined in six different hospitals in Morocco. The data were collected from CT Brain examinations at of 50 CT scans per hospital. For each examination, we have reported CT acquisition parameters including the number of series, use of contrast medium; tube kV, tube current and rotation time, slice thickness as well as the displayed CT dose index ($CTDI_{vol}$) and the Dose Length Product (DLP). Cancer and biological heredity risks were estimated using the International Commission on Radiological Protection (ICRP) conversion factors.

Results: The mean effective dose received by children during brain CT scan was varied between 0.76 and 7.79 mSv with an average value of 2.69 mSv. The patient cancer risk per procedure ranged between 4 and 43 per 10^5 CT procedures with an average value of 15 per 10^5 CT procedures. The patient heredity risk varied between 2 and 16 per million CT procedures with an average value of 5 per million CT procedures. These results proved that the patients were exposed to needless doses of radiation during their CT scans.

Conclusion: The mean effective dose obtained was higher than those established in some countries. A wide variety of doses per scan were seen between the 6 hospitals, and differences were noted between patients even within the same facility. This study showed the need for practice unification and radiation dose optimization during CT scans of pediatric patients.

Keywords: *Brain CT scan, Children, Radiation doses*

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PS2 (T2.1-1243)

DEVELOPING NATIONAL BIODOSIMETRY CAPABILITIES FOR PREPAREDNESS AND RESPONSE TO NUCLEAR AND RADIOLOGICAL EMERGENCIES IN THE NATIONAL NUCLEAR CENTER OF KAZAKHSTAN

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Nuclear and other tests carried out at the Semipalatinsk test site, designed to establish unique, long-term conditions for dose formation among a large number of residents and their generations from nearby territories. It does not seem possible to reliably and precisely find the individual exposure values (energy and composition of radiation, dynamics of dose accumulation and so on) after all the time have passed – over 70 years since the tests were started. Despite the fact, that after the STS was closed, a lot of researchers used different dosimetry techniques independently attempting to estimate radiation exposures in inhabited localities in various places close to the former Semipalatinsk Test Site and to determine effective equivalent doses, findings are highly contradictory and differ from one another significantly.

Republic of Kazakhstan is the second largest country in the world in uranium inventory and remains a leader in its production (39% of the world's inventory). KazAtomProm includes 80 subordinate organizations with about 27,000 employees. Our country is planning to build the first nuclear power station, has spent nuclear fuel and radioactive waste disposal projects. Scientific research activities are actively developed in the field of nuclear energy, radiation sources are increasingly used in diagnostic, therapeutic and medical procedures.

Our institute became proficient in cytogenetic biodosimetry methods such as dicentric analysis of, cytokinesis-block micronucleus assay, fluorescence in situ hybridization (FISH) method. Availability of such unique in Kazakhstan highly technical equipment as Metafer 4, ICAROS, ISIS, MN Score software (MetaSystems, Germany) in the Institute of Radiation Safety and Ecology NNC RK significantly contributes to good quality reliable indication of absorbed dose.

Retrospective assessment of individual doses received and indication of stable translocations made using FISH method, by means of whole-chromosome probes for 1,4,12 chromosomes (MetaSystems, Germany). This research is the most interesting one, taking into account historical radiation history of our region. We researched a group of 37 aboriginal inhabitants of East-Kazakhstan region, not suffering professional exposure. The group representatives ranged by gender, age and smoking bad habit. Background frequency of stable translocations in different age groups differs between 0,97±0,1 and 6,37±0,8 per 1000 cells.

Since Kazakhstan is the 9th largest country in the world, and taking into account formation of natural-artificial provinces as the result of business activities, it's reasonable to determine regional background spontaneous frequency of non-stable aberrations in other regions of our country: Norther, Southern, Western and Eastern. In our opinion, indication of received dose should be assessed using regional coefficient of values variation for each group of people. Results of researches show low enough values of frequencies, that can be compared with general population data and bookish data from other researchers. Observed results of researches demonstrate interregional variability within the range of 1,1±0,4 to 3,09±0,6 aberrant cells per 1000, which is caused by several subjective and objective factors of life activity.

Individualized dose quantification using cytogenetic biodosimetry will ensure the reliability of decisions regarding the response and mitigation of health consequences in the event of unforeseen radiation accidents. It will also reduce the chronic social tension associated with radiophobia, stop speculation and legal battles associated with diseases from radiation, and allow clarifying the radiological situation in the territories adjacent to the STS.

From 2020 to 2023, we are implementing a Project entitled "New Biological and Physical Triage Methods in Radiological and Nuclear (R / N) Emergencies" (BioPhyMeTRE) endorsed under the NATO Science for Peace and Security (SPS) Program. Coordinated by the Italian National Agency for New Technologies, ENEA of Italy, the Institute for Radiation Safety and Ecology (IRSE) of the National Nuclear Center of Kazakhstan, the National Institute of Health (ISS) Italy and the Ruger Boskovic Institute (RBI) of Croatia you can find out on the Project website https://biophymetre.com/

PS2 (T2.1-1246)

The measurement of accidental radiation doses by Fukushima Dai-ichi Nuclear Power Plant Accident: ESR dosimetry of cattle tooth enamel

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Radionuclides were released to the environment by the Fukushima Dai-ichi Nuclear Power Plant accident. It is an important issue to estimate the radiation doses to animals in order to know the impact of the accident to the environment. In the present study, we employed ESR dosimetry to apply to tooth enamel of cattle, the method of which has already been established for human tooth enamel. Stable CO_2^- radicals, to be detected by ESR, are created in tooth enamel made of hydroxyapatite by radiation, being accumulated through life.

Tooth samples were extracted from the jaw of the cattle. We used two teeth of each cattle. Total of 20 Japanese cattle from 4 to 12 years old raised in ranches in Okuma-Ikeda and Namie-Omaru which are located in contaminated areas. The teeth were removed from the jaws and cut in halves with a saw. Dentin was removed by dental drills, and soaked in a 20% KOH solution at 60°C with ultrasonic. After being washed and dried, the samples were crushed to less than 1 mm with sieves. The ESR measurements were performed for the samples, and the intensities of the dosimetric CO_2^- radical signal were obtained by a computer program to extract that component from the spectrum.

The accidental radiation doses were calculated by dividing the signal intensity obtained from the ESR spectrum by the sensitivity, which is the slope of the line fitted to the dose response of the cattle tooth enamel. The maximum of the obtained doses was about 1.2Gy. These dose values were then compared with the integrated environmental dose rate monitored at Omaru ranch where the changes of the dose rate with time were integrated for the period from the formation of the cattle tooth enamel until the cattle was provided for examination. As results, these estimated values are roughly consistent with the those obtained by the ESR measurements for the samples of the Omaru ranch. As for the samples of the Okuma ranch, there are samples, the obtained doses of which are below the detection limit, while some show high values that do not match the integrated values. All latter cattle have records of moving from other ranches. Most probably, these cattle were exposed to radiation before moving. We also investigated the difference in absorbed doses corresponding to the difference in the eruption time depending on the tooth position. In cows with all permanent teeth erupted before the earthquake, there was no difference depending on the position of the teeth, but in young cows, the dose of M1 that erupted earlier was higher than that of P4.

The accidental doses were practically detected from the teeth of the cattle, and the values were close to the estimated doses of the ranch obtained from the air dose rate. The difference in doses was also detected corresponding to the difference in the time of tooth eruption. We conclude that ESR dosimetry with cattle teeth practically works well.

Poster Session T2

PS2 (T2.2-0621)

Simulations on the absorbed dose conversion coefficients based on a shark voxel phantom

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Abstract: For determination on the absorbed dose of the sharks living in the ocean, the dose conversion coefficients with adult sharp was derived by an adult shark voxel phantom and monte Carlo simulation. The adult shark was captured on 2013 with length 1.75 m and mass 20.64kg. The shark was scanning by a computed tomography (CT) with 512 (pixels) * 512 (pixels) * 208. based on this original CT images, the shark was segmented in to 17 organs, and each voxel size is 4.88mm * 3.904mm * 10mm. This voxel phantom was loaded in a GEANT4 code. The conversion factors were calculated with Geant4 for both the external radiations and the internal radiations. The external radiative sources include the ^{134}Cs, ^{137}Cs and ^{40}K which are the main radioactive components of the seawater polluted by the liquid release from the nuclear power plants or nuclear emergency. While the internal radiations mainly come from the ^{134}Cs, ^{137}Cs and ^{110m}Ag when the sharks are living in the contaminated sea water. The conversion coefficients from source organs to target organs of different radionuclides are calculated, respectively. These conversion coefficients benefit the evaluations on the absorbed dose received by the shark living in the polluted waters and can be used on the assessment of the environmental influences on the shark due to the liquid release of the nuclear power plants.

keywords: absorbed dose; shark phantom; voxel phantom

Poster Session T2

PS2 (T2.2-0675)

DETERMINATION OF ALPHA AND BETA RADIATION DOSES TO THE SKIN OF POTTERS DUE TO THE NUCLEI OF THE ^{238}U AND ^{232}TH RADIOACTIVE SERIES FROM THE APPLICATION OF CLAY

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Uranium (^{238}U), thorium (^{232}Th), radon (^{222}Rn) and thoron (^{220}Rn) contents were measured in various clay samples used for pottery production in different regions in Morocco by using two types of solid state nuclear track detectors. Radiation doses to the skin of potters due to alpha-particles emitted by the ^{238}U and ^{232}Th radioactive series were evaluated. Estimates of the annual committed equivalent doses to the skin of potters due to the nuclei of the ^{238}U and ^{232}Th series ranged between 8.0 mSy y^{-1} cm^{-2} and 15.13 mSy y^{-1} cm^{-2} and between 2.14 mSy y^{-1} cm^{-2} and 4.16 mSy y^{-1} cm^{-2}, respectively. Committed equivalent doses to the skin of potters due to beta particles emitted by the nuclei of the ^{238}U and ^{232}Th series inside the studied clay samples were determined.

Keywords : Nuclear Track Detectors, Clay Samples, ^{238}U and ^{232}Th Concentrations, Radiation Dose Assessment to Skin.

PS2 (T2.2-0708)

Investigations of effects on intensity of radiation-induced signal with tooth enamel dimensions *for in-vivo* EPR tooth dosimetry

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In vivo EPR spectrometer has been developed as a device used at the first stage of triage to meet the requirements for swift response to large scale radiation emergency in limited infrastructure and human resources [1]. However, the sensitivity of measured signal intensity could be affected by tooth enamel dimensions when the consistent delivered dose. In this study, we investigated the variations of *RIS* (Radiation-induced signal) with aspect ratio (AP) and enamel thickness (ET) using electromagnetic simulation software (ANSYS HFSS software, ver. 18) based on finite element analysis. The resonator of *in vivo* EPR spectrometer was modeled in the simulations. This resonator consisted 10 mm loop using 1.2 mm thick silver wire. *sRIS* (simulated *RIS*) could be calculated using HFSS based on the stored magnetic energy within the enamel box and characteristic of the resonator assuming the constant incidence microwave power [2]. The *sRIS* was calculated to enamel box with designated enamel volume (EV) instead of tooth by multiplying the ET and surface area. ET was measured from the Micro-CT images at the center of the tooth (20 human incisors). The calculated *sRIS* were normalized at those of enamel box with EV of 93.3 mm^3. The linear relationships between EV and normalized *sRIS* according to ET and AP were described in Fig.1a-b. However, this linearity was worse in thin enamel box (Fig.1a) and relatively larger enamel box (Fig.1b). The *sRIS* for each simulation set was corrected by dividing the EV of each enamel box for evaluation when applying geometry correction. The corrected *sRIS* for the simulation set was consistent to less than 3.9% (Fig. 1c) and 2.8% (Fig. 1d) in the range of EV from 70 to 120 mm^3. Based on the findings, ET affects *RIS* that might induce significant discrepancy rather than AP when using the geometry correction. The geometry correction should carefully apply to larger or smaller EV of tooth in vivo EPR tooth dosimetry.

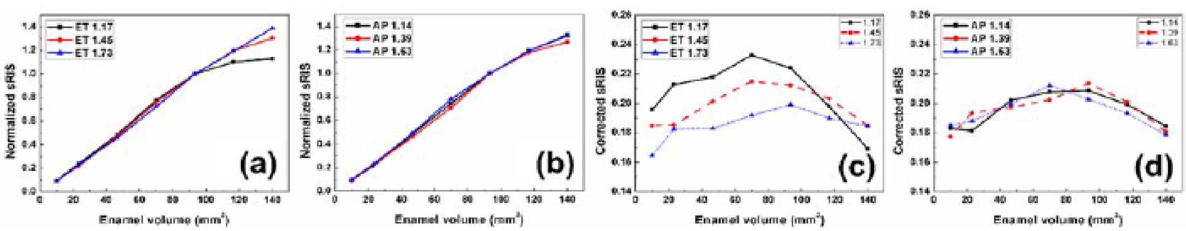

Fig. 1. Results were calculated from seven simulation sets with seven equidistant intervals from 10 to 140 mm^3 for EV. The variations of normalized *sRIS* with (a) ET multiplying the median value of AP (1.28), and (b) AP multiplying the median value of ET (1.73 mm). The variations of corrected *sRIS* with (c) ET, and (d) AP.

Keywords: *tooth dosimetry, in vivo dosimetry, electron paramagnetic resonance*

ACKNOWLEDGMENTS

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PS2 (T2.2-0721)

Evaluating the Radiological dose Using The RESRAD code

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The purposes of this study are using the RESRAD-OFFSITE computer code for the radiological safety assessment of decommissioning nuclear power plants, the analysis of radiation exposure pathways, and the simulation of residual radioactive contamination. This study can provide the process of simulation, the design of parameters, the result of analysis, and the radiological safety assessment of humans and environments using the RESRAD-OFFSITE computer code. The RESRAD family of codes is developed by US Argonne National Laboratory (ANL) to analyze potential human and biota radiation exposures from the environmental contamination of residual radioactive materials. The RESRAD family of codes is widely used by regulatory agencies, the risk assessment community, and universities in the world. The results of this study can also provide the simulation methodologies and verification techniques with using the RESRAD-OFFSITE computer code for the radioactive contaminated buildings, the decommissioning nuclear power plants, and the radioactive waste storage facilities. We derivated the DCGL (Derived Concentration Guideline Level) with License Termination Plan (LTP) of Maine Yankee using RESRAD with four element including Cs-137, Co-60, H-3 and Ni-63 . For the initial concentration of soil nuclides in the contaminated area were referred to the Maine Yankee LTP data of the US decommissioned NPP (Maine Yankee Atomic Power Company 2005). This type of nuclides analysis is shown in Table 1. There were some successful decommissioning cases in the United States. Most of these decommissioning cases use similar parameter selection processes for DCGL calculations, and their calculations were also officially approved by the US NRC. We also use the RESRAD code to calculate the DSR and use the formula to find out the DCGL.

Table 1. Initial soil concentration

Nuclides	Half-life (years)	Concentration (pCi/g)	Proportion (%)
Cs-137	30.2	4.3	89
Co-60	5.3	0.0434	0.89
H-3	12.4	0.256	5.3
Ni-63	100.1	0.232	4.8

***Keywords**: Decommission,RESRAD,DCGL*

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PS2 (T2.2-0748)

Generic study on radiation exposure of medical staff caused by radiological interventions

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Since many years, the importance of interventional measures is increasing. During these medical applications X-ray tubes are used while medical staff is present and partly near by the primary beam. Therefore, additional radiation protection equipment is applied in order to reduce the radiation exposure of the medical staff. Nevertheless, the increasing number of interventions raises the question if the limits for occupational exposure can be exhausted and which radiation protection measures are necessary to prevent exceeding the legal limits. To investigate the radiation exposure of the medical staff this research project has been initiated. Experiments using phantoms and static as well as well-defined parameters have been performed at the University Hospital Augsburg and University Hospital Cologne. The experimental results were intended to use for validation of a Monte-Carlo simulation which can reproduce the measured values. In addition, the simulations should allow for predictions of values that are beyond experimental accessibilities. By means of such a simulation environment, complex problems can be investigated, and radiation protection can be optimized.

PS2 (T2.2-0771)

Potential radiation burden after the death of patients treated with radioactive substances

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The number of therapeutic nuclear medicine procedures is expected to increase, with a wider range of radiopharmaceuticals. It is possible a patient may die shortly after the treatment with radioactive substances, but the radiological risks of the subsequent burial or cremation are not always well known. This study aims to estimate the radiological risks associated with the death of patients treated with radioactive substances.

For this work, several nuclear medicine therapies were selected based on their radiotoxicity and their frequency of use in the Netherlands. Exposure scenarios were constructed for representative persons involved in funerals, based on interviews and site visits to funeral homes. For these representative persons, effective and local equivalent skin doses were modelled per event. Both external and internal exposure pathways were taken into account.

We found that family and close friends of the deceased were exposed to the highest effective dose, in particular for the burial of patients treated for thyroid carcinoma with iodine-131 (1.8 mSv). For that treatment, we also found the highest equivalent skin dose (180 mSv for crematory staff). The effective dose of residents living near a crematory did not exceed 10 μSv in any scenario. We will discuss these and further results and their underlying methods.

The Dutch authorities are using this study to evaluate guidelines for nuclear therapy and the funeral industry.

Keywords*: Funeral, cremation, radiopharmaceuticals*

ACKNOWLEDGMENTS
This work was financially supported by the Dutch Authority for Nuclear Safety and Radiation Protection (ANVS).

PS2 (T2.2-0790)

Demonstration the effectiveness of Monte Carlo-based data sets with the point-source light-of-sight approximation for shielding design of a laboratory with the therapeutic level proton beam

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There are several proton therapy facilities in operation or planned in Taiwan, and basic research facilities with hundreds of MeV-level proton beams are anticipated to increase. Monte Carlo simulations are generally considered the most accurate method for complex shielding analysis; while, it is time-consuming, and requiring knowledge and experience when dealing with a challenging shielding problem including complicated geometry and thick shielding. In contrast to accurate Monte Carlo simulations, simplified approaches based on the point-source line-of-sight approximation are friendly and often preferable in the initial design phase because they are intuitive and easy to use. In principle, one can quickly estimate the transmitted dose rate at any position outside the shielding by selecting the shielding parameters base on the problem. From literature, most of the comparisons have revealed considerable differences in predicted doses because of various uncertainties, such as dose conversion factors, beam-target configurations, methods for solving radiation transport, and underlying cross-sections. To demonstrate the effectiveness of the Monte Carlo-based data sets, this study emphasized a consistent comparison between the data sets with the point-source light-of-sight approximation and those acquired by Monte Carlo simulations for dose rate around the laboratory with the therapeutic level proton beam.

Three calculations were conducted with the MCNP Monte Carlo code in this work. First of all was neutron yield calculation by direct modeling of a similar proton bombardment experiment to confirm the selected physical model used in the followed calculation. Secondly, simulation of the radiation transport in a simplified spherical geometry used to generate a set of shielding data, including source terms and attenuation lengths for several beam-target-shielding configurations, especially for PMMA shielding material. The last calculation was used MCNP to simulate the radiation transport in a practical case study, including a beam dump made of graphite, iron and PMMA.

The comparison results indicate that the Monte Carlo-based data sets with the point-source light-of-sight approximation provide reasonable estimates of the transmitted dose rates, but a fly in the ointment was that the doses near maze entrance were underestimated. This is inevitable, because of the dose at the entrance of the maze consists of radiation transmitted part and the radiation streaming part. In this regard, the Monte Carlo-based source terms can be used in estimating the doses along the maze by a proper coupling with semi-empirical formulae that parameterize the behavior of neutrons streaming through labyrinths.

This case study proved the reliability of the data set in conducting dose evaluations for the laboratory with the therapeutic level proton beam, but having said that, caution should be taken that verifying the results by using independent methods before finalizing any shielding design is necessary. Because the shielding parameters are generated using simplified geometries that may cause non-negligible effects of the actual geometry on radiation transport to be overlooked.

Keywords: *Monte Carlo; point-source light-of-sight approximation; source terms and attenuation lengths*

PS2 (T2.2-0801)

Development of a Computer Program to Calculate Radiation Dose Due to Radon From Consumer Products

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Radon, one of natural radionuclides, is a colorless, odorless radioactive gas. Because radon is an inert gas, it easily travels within the material and is released into the air from the material surface. Recently, there have been cases in which radon is released from consumer products due to their precursors in some consumer products. Radon released from consumer products increases the concentration of radon in indoor air. Internal radiation exposure can occur due to the inhalation of the radon. Social concerns have been raised recently. A computer program was developed to calculate internal radiation dose due to the radon emitted from consumer products. To develop the computer program, we considered many dose influencing factors such as radon concentration, equilibrium factor, breathing rate, exposure time, and dose coefficient. The major exposure factors suggested by ICRP and UNSCEAR were used as input data. IAEA and ANL suggested methods for evaluating radon concentration in indoor air in consideration of radon emanation and diffusion[1, 2]. Using the methods, indoor radon concentration can be estimated based on radium concentration and the physical properties of the NORM material. The methods use radon exhalation rate to estimate the radon concentration in indoor air. The exhalation of radon occurs through the emanation and diffusion process. For calculation of radon emanation, the emanation coefficient and moisture content of the material are considered. For calculation of radon diffusion, the diffusion coefficient, the porosity of the material, and the moisture content are considered. For calculation of radon exhalation, the concentration of radium, density of material, emanation coefficient, and diffusion coefficient of radon are considered. If the measured value of radon concentration is not available, the program can estimate and evaluate the radiation dose using the corresponding algorithm. However, the algorithm does not consider the daily and seasonal changes in radon concentration. The program can be used as a basis for establishing a methodology and estimating radiation dose for radon released from consumer products.

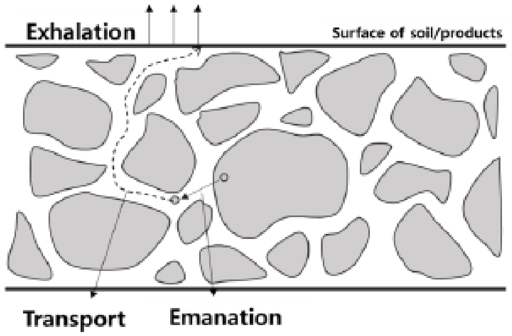

Fig. 1. The schematic for 3 steps of radon release

Keywords: Dose assessment, Radon, Consumer products

ACKNOWLEDGMENTS
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PS2 (T2.2-0816)

Analysis of counting efficiencies for whole body measurement depending on the individual characteristics

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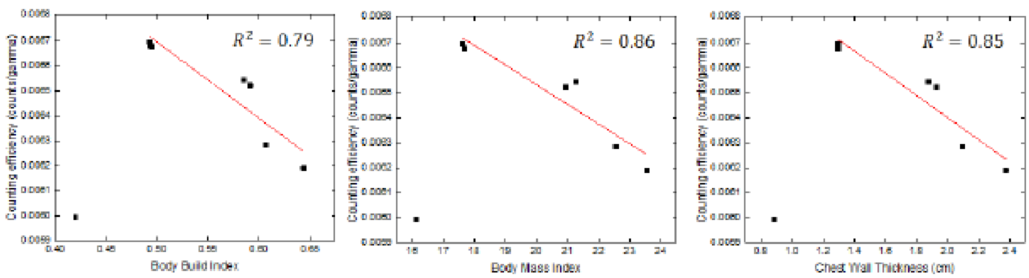
It is necessary to perform the proper calibration for obtaining the accurate internal dose assessment, but there are some limitations in conventional calibration procedure. The Bottle Manikin Absorber (BOMAB) physical phantom is widely used for calibrating the whole body counter (WBC) but the geometrical discrepancies between physical phantom and the real human body can influence the measurement results of internal contamination [1]. Therefore this study aims to analyze the dependency of efficiencies, considering the individual characteristic in the whole body measurement.

Monte Carlo simulation was applied to calculate the whole body counting efficiencies depending on the individual characteristics including the age and gender. Stand-type WBCs are modeled with a Mote Carlo code for simulation. ICRP adult reference computational phantoms and the UF/NCI reference hybrid family phantoms [2] were used to represent the anatomical characteristics. The efficiencies for male and female of 5, 10, 15y and adult were obtained using those computational phantoms.

To find the tendency of counting efficiencies considering the individual characteristics, some of the indexes such as the body mass index (BMI), body build index (BBI) and chest wall thickness (CWT) were used for quantifying the anatomical characteristic. As shown in Fig 1, the counting efficiencies were decreased in accordance with the increment of physical indexes. The counting efficiencies of 5y phantoms were smaller than others, because the stand-type WBC had static measurement geometry. Thus, the counting efficiencies of 5y phantoms were not included in fitting equations. The BMI and CWT highly correlated with counting efficiencies and it can be possible to estimate the individual counting efficiency based on this relationship.

The anatomical characteristics depending on the age and gender of individual can influence the counting efficiencies of whole body measurement. Thus, it is necessary to establish a whole body measurement methodology with using the relationship between the counting efficiencies and the individual characteristics.



(a) (b) (c)

Fig. 1. The variation in counting efficiencies for 661 keV from Cs-137 as a function of physical indexes: (a) Body build index, (b) Body mass index and (c) Chest wall thickness

***Keywords**: Whole body measurement, Monte Carlo simulation, Computational phantom*

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PS2 (T2.2-0850)

Theoretical simulation and experimental study on material activation for the key components of NBI

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The neutron source intensity in the plasma center is on the order of 1E+20n/s by DT fusion reaction, during the China Fusion Engineering Test Reactor (CFETR) fusion power operating with 1.5GW. Fusion neutrons stream easily into the interior of the neutral beam (NB) modules through the wide ports and the drift ducts. Since there are no obstacles in the neutral beam path, a very high neutron flux penetrated the beamline and activate the NB module. The shutdown dose rate caused by activation will impact for the neutral beam injection's maintenance man. Also, the material stability will be effected by the decay heat released from the material activation. So it is important to calculate the material activation for the neutral beam injection (NBI).In this paper, the Monte Carlo method and the two-dimensional RZ geometry model are used to analyze the neutron fluxes and the materials activation for NBI. The results demonstrate that the induced radioactivity of the Calorimeter (CAL), the Residual Ion Dump (RID) and the support components of the CAL and RID are Cu and SS316, during the CFETR operation. And then, the activation samples were designed in order to observe the activation results experimentally. Then, based on the HINEG device, the copper sample is irradiated 20 minutes by DT neutron source. Next, we measure the gamma-ray spectra of the samples using the high purity germanium detector. Theoretical simulation and experimental measurement results are shown in Fig. 1 and Fig. 2.

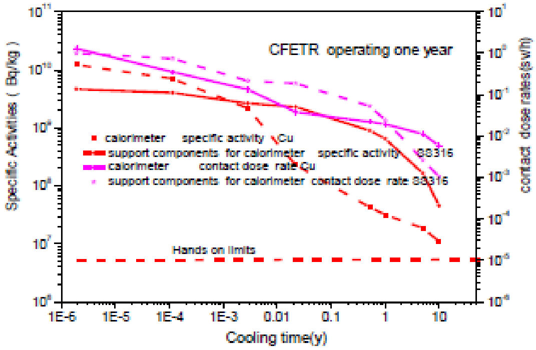

Fig.1 The specific activity and contact dose rates

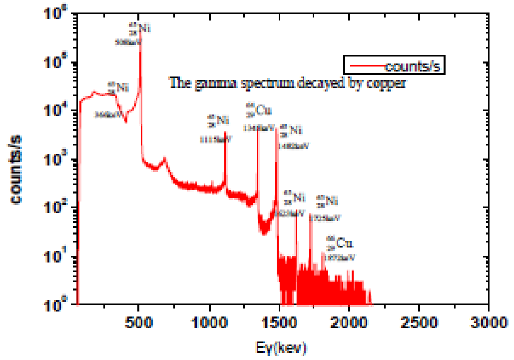

Fig.2 The gamma spectrum decayed by copper.

Keywords: NBI, Material activation, Montle Carlo method

ACKNOWLEDGMENTS

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PS2 (T2.2-0855)

Estimation of Eye Lens Dose In Scattered Radiation Fields

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Objective In decommissioning of the nuclear power plants after an accident such as the TEPCO Fukushima Daiichi Nuclear Power Plants (1F), situations are different from that under normal operations of nuclear facilities. High dose γ-ray fields due to fission products released from reactor buildings may exist after decontamination and shielding of these radiation exposure sources. The radiation workers would also be exposed to scattered radiations in these fields. Now Japanese regulation authorities are preparing to revise the dose limit to the lens of the eye for occupational exposure in planed exposure situations. The revised dose limit to the lens of the eye will be the same as the lens dose limit recommended in the ICRP statement (100 mSv/5 years and 50 mSv/year). Thus, it is necessary to discuss, whether the high dose radiation workers comply with the regulations related to the dose limit to the lens of the eye, and how to measure radiation dose of the lens. In this study, the equivalent dose of the lens for the decomissioning workers in gamma-ray fields including the scattered radiation was estimated by calculation.

Methods The equivalent dose of the lens and dose quantities ($H_p(10)$, $H_p(3)$ and $H_p(0.07)$) used to estimate the equivalent dose of the lens were calculated by the Mont Carlo simulation code, PHITS. The equivalent dose of the lens was estimated using the detailed eye model placed in the MIRD phantom shown in ICRP Publication 116. And these values are compared to $H_p(10)$, $H_p(3)$ and $H_p(0.07)$ estimated by simulating an ideal dosimeters worn on the surface of the head, the neck, the chest and the upper arm of the MIRD phantom inside and outside the shielding garment. A gamma radiation source was prepared based on the results measured by a gamma-ray spectrometer around the 1F Unit 2 Reactor.

Results As operational doses calculated for Antero-Posterior (AP) and Rotational irradiation geometries in gamma-ray field where the contribution of scattered radiations to the dose was higher than ^{134}Cs and ^{137}Cs, $H_p(10)$, $H_p(3)$ and $H_p(0.07)$ calculated for ideal dosimeters set on MIRD phantom outside the shielding garment on were higher than equivalent doses of the lens calculated for the detailed eye model. However, the value of $H_p(3)$ calculated at the position of the chest inside the shielding garment was almost half of the calculated equivalent dose of the lens. This means that the equivalent dose of the lens may be underestimated when the equivalent dose of the lens was estimated using the value measured by a personal dosimeter worn on the chest inside the shielding garment. In addition, for AP irradiation geometry, $H_p(10)$, $H_p(3)$ and $H_p(0.07)$ calculated for the area set in the back part of upper arm outside the shielding garment was lower than that at other parts of the upper arm. This is because that the torso and shielding garment block the incidence of gamma-rays on this area.

Conclusion
We developed calculation method to estimate the equivalent dose of the lens and concluded that these calculated results could complement the lens dose results measured by the personal dosimeters.

Keywords: *Equivalent dose of the lens of the eye, ICRP detailed eye model, scattered radiation field,*

Poster Session T2

PS2 (T2.2-0904)

A Comparison of Two Methods for Modeling the Biological Distribution of Radioactive Progeny Born inside the Body

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A unique problem arises when applying biokinetic models for radionuclides which are parents of decay chains. Since radioactive progeny usually behave differently from their parents, it is often the case that progeny and parent will have different tissues specifically invoked in their respective systemic biokinetic models. These systemic models typically have an 'Other' tissue compartment which contains any systemic activity not assigned to an explicitly named compartment in the model. Since the set of explicitly named compartments may vary throughout the chain, the definition of 'Other' varies. The solution to the problem requires a method be invoked to compute the activity in a common set of tissues for all members of the chain. ICRP Publication 71 describes two such methods but suggests they may yield different results. The publication further states that the differences are small as long as the mass of 'Other' is large. This work provides a description and comparison of the two methods. Using simple biokinetic models for a two-member chain, mathematics is used to prove the equivalency of the two methods. Further, numerical solutions of both the simple model and actual biokinetic models are given to show equivalency of the methods. As long as the methods are applied utilizing the mass of 'Other' for the appropriate chain member, the two methods are equivalent for handling the in-growth of radioactive progeny in systemic biokinetic models.

PS2 (T2.2-0923)

Estimating real-time measurement doses for eyes in interventionist procedures

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The interventional medical procedure, the medical staff, is subject to high exposure to scattered radiation when compared to the other occupational groups working with X-rays. Of course, radiation protection is a matter of interest to the interventional medical team. Recently, the International Commission on Radiological Protection recommended decreasing the occupational dose in the lens of the eye from 150 mSv to 20 mSv annually, indicating that dose limit accumulated in the lens of the eye to avoid a relative risk of developing a cataract after absorbed by the lens of the eyes. For the protection of the radiation is used of a dosimeter for the control of the accumulated dose and are verified after a long period for the analysis of the radiation. The dosimeters are radiation measuring instruments and are usually placed on the physician's chest while reading the dose of radiation received on the eye lenses is compromised by the accuracy of the measurements. The objective of this work is to develop a system that allows estimating the dose accumulated in real time in the lens of the eye received by the physician during the interventional procedure and indicate means of its protection against ionizing radiation. The performance of the remote-controlled fluoroscopy system in scattered radiation measurements was tested at 50 kV and 30 mA with a Ludlum 9DP ionization chamber and a PMMA phantom. Dose rates were collected by the ionization chamber at 166 cm about the ground level, with distances of the main beam at 19, 38, 76, and 152 cm, around the fluoroscopic tabletop at 45 degrees and 45 degrees. To estimate equivalent doses in real time, an ionization chamber was attached to the top of the fluoroscopic tabletop and used as a fixed reference point. The collected data were sent to the developed software to determine the estimated doses. The received data were transmitted to the software designed to assess the equivalent dose estimates. The calculated equivalent doses were visualized in real-time through a display showing the intensities of the accumulated dose with a visual indication of the color scale for the doses received by the members of the medical staff. An interventional procedure has an average duration of 13 minutes. To test system performance, we used two phantoms with different heights to measure the equivalent dose estimate. We radiate the phantom separately, and we found, respectively, the following ratings, in the 25x25x7 cm phantom: 76.58, 35.09, 23.96 and 0.27 μSv, and another of 30x30x15 cm: 686.17, 318.61, 108.5 and 2.45μSv, respectively measured in real time at a height of 166 cm, in the region of lens of the eyes. The results show the dose intensities in the system viewer the doses received to the distance the physician is from the main beam. The estimated dose in real-time in relation to the distance helps the interventionist physician optimize their exposure to scattered radiation during the intervention procedure, allowing them to enhance personal protection and reduce occupational dose in the lens of the eyes.

Keywords: *Eye lens dose; Real-time measurement; Interventional radiology; Monitoring practical aspect.*

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PS2 (T2.2-0932)

Reconstruction of Historical Radiation Doses in a Cohort of Korean Radiation Workers

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Radiation workers in Korea have been monitored for their official occupational radiation doses since 1984. The dose was obtained from the Central Registry for Radiation Worker Information (CRRWI) managed by the Nuclear Safety and Security Commission (NSSC). The purpose of this study was to estimate the historical radiation doses for workers with missing or deficient exposure information. The Korean radiation workers cohort included 20,608 workers, and their radiation doses are available since 1984. The historical reconstruction models were developed using cross-sectional survey data and the personal badge doses. The overall model included calendar year, age, and facility type and the models were also constructed in six categories according to their sex and facility types (e.g., Nuclear power plant, Industrial radiography and others). The comparison of the measured mean doses from badge doses and estimated mean doses from the reconstruction models were shown in Fig 1. The estimated mean doses decreased from 3.35 mSv in 1984 to 0.72 mSv in 2016, showing a similar trend of the measured doses from 4.50 mSv to 0.60 mSv. Due to the insufficient data of workers in early period of the dose registry, there was relatively large uncertainty in estimating radiation doses of workers with early employment. Further studies are needed, incorporating additional exposure-related occupational characteristics into the reconstruction models.

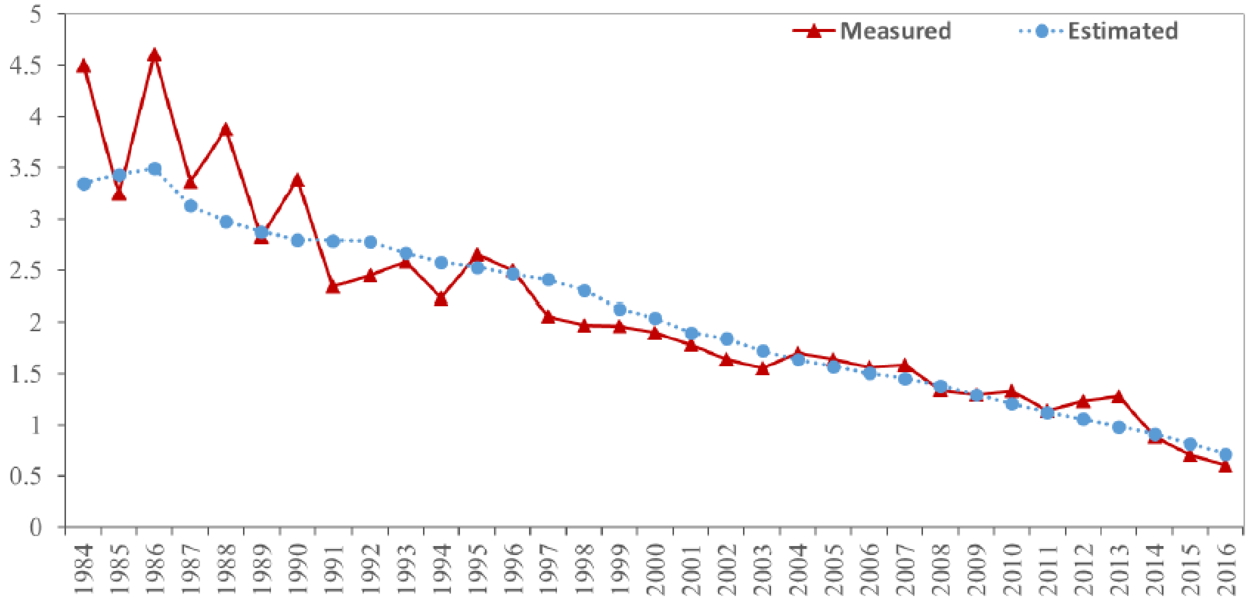

Fig. 1. Comparison of mean estimated doses with mean measured doses for radiation workers, 1984-2016

Keywords: *Dose reconstruction, Occupational exposure, Radiation workers*

PS2 (T2.2-0955)

Monte Carlo modeling of neutron and gamma dose rate to select microorganisms *in situ* from a laboratory plutonium-beryllium source

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Monte Carlo N-Particle (MCNP) transport code was used to model dose rate from a 37 GBq plutonium-beryllium (^{239}PuBe) alpha-neutron source to biological samples *in situ*. The objective of this work was to characterize the dose rate to representative microorganisms (e.g., bacteria and yeast) in order to develop neutron dose-response models for these organisms, accounting for coincident dose from gamma irradiation. In the experimental setup, a ^{239}PuBe source is contained within a neutron howitzer (Fig. 1), which is a large drum filled with paraffin wax that has a vertical source port and four horizontal irradiation ports intended for use in laboratory-based neutron experiments. To expose the microorganisms in this fairly unique geometry, cells were grown on a polycarbonate filter on top of a minimal medium 35×10 mm agar plate. A breathable film was placed over the plate, which was then placed vertically in an irradiation port with the filter side facing the source for 48 hours. Initially, foil activation was used at various distances from the source in each irradiation port to determine thermal neutron flux, which was then used to verify the Monte Carlo model representing the system. The model was then adapted and refined for dosimetric modeling. Consideration was also given to the influence of variability in the samples, such as the thickness of the agar and small shifts in distance from the source, on the dose rate.

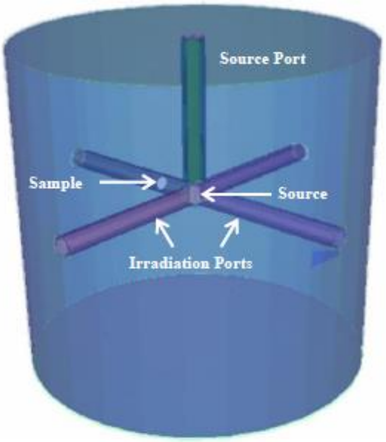

Fig. 1. Modelled neutron howitzer with four irradiation ports and representative sample.

Keywords: *neutron, dosimetry, Monte Carlo*

ACKNOWLEDGMENTS

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PS2 (T2.2-0962)

The importance of the computational phantom posture in neutron organ dose assessment

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Mesh-type reference computational phantoms (MCRPs) are the evolution of the previous reference voxel-type computational phantoms (VCRPs) as they are a more detailed description of the human body, thus overcoming the limitations of the latter [1]. Because thin and complex organs and tissues are realistically described in the MCRPs, it is possible to achieve a more accurate dose calculation in these structures. The MCRPs were created in different postures, such as squatting, bending and standing up, to account with the different positions a person assumes when dealing with different scenarios. These postures realistically represent a person when setting up experiments and walking or standing by or next a radiation source.

To understand how the different postures affect the dose delivered to different organs, these were implemented in front of the model of a neutron Howitzer container using the Monte Carlo code PHITS (Figure 1). This model corresponds to the neutron Howitzer container at the Neutron Measurements Laboratory of the Energy Engineering Department of the Polytechnic University of Madrid (UPM) equipped with a ^{241}Am-Be neutron source in its center. The container allows the source to be in either the irradiation or the storage position [2]. The neutron and gamma dose was calculated in several key organs, such as the eye, eye lens, kidneys, thyroid, among other organs, using the aforementioned postures. The results obtained with the different MCRP postures were compared with each other and also with the Golem voxel phantom and the significance of using the different postures is discussed.

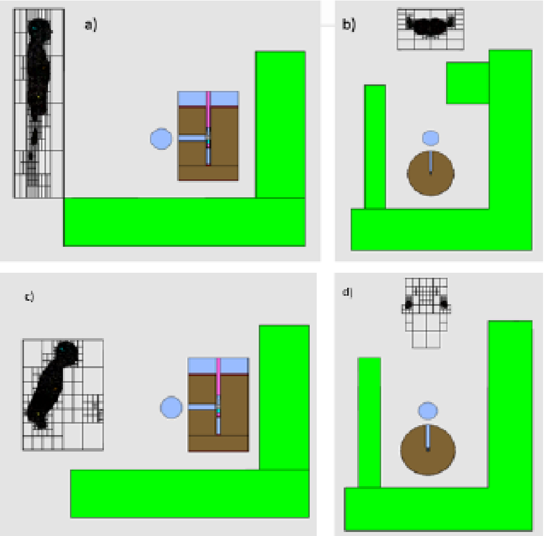

Fig. 1. Standing (top) and squatting (bottom) MCRP postures.

Keywords: MCRP, VCRP, Organ dose

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PS2 (T2.2-0974)

Mcnpx Simulation Method for Extremity Dosimetry of 3D Scanned Hand Model

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It is often necessary to perform MCNPX simulations for objects with free curved surfaces when performing a dose estimation with respect to radiation exposure incidents and a precise design at inhomogeneous radiation fields. However, in general, there are two obstacles when it comes to performing such simulations. One is to measure or imitate objects with free curved surfaces, and the other is how to define simulation input mathematically.

In the first place, it is not easy to create a model that imitates the shape and motion characteristics of, for example, hands locally exposed to radiation, as that would require countless detailed measurements. Even though the exposed fingers can be simplified into a cylindrical shape and phantoms are available, these cylindrical fingers do not reflect the original shape, and voxel and polygon phantoms are made from a still human body and thus cannot emulate the different motion features accurately and easily. In this paper, we suggest a simpler and more accurate modeling method based that directly uses 3D scanning to imitate targets, although there is an error in dose calculation due to the absence of the bone.

Secondly, even though voxelized models have been employed for MCNPX simulations of objects with arbitrarily curved surfaces, the voxelization process tends to distort the polygon models obtained from the 3D scan of targets, making simulations of thin objects difficult and generating artifacts. For this reason, although polygon phantoms are suitable for arbitrarily curved surfaces, they have been mainly developed for the Geant system. Therefore, the second goal of this research is to develop a tetrahedralization method for mathematically defining measurements on MCNPX.

We present a simulation method that can create accurate virtual models of hand with arbitrarily curved surfaces and perform distortion-free MCNPX simulations. Generally, MCNPX simulations of objects with arbitrarily curved surfaces are performed through voxelization. In this study, a polygon model is tetrahedralized by TetGen for the construction of the MCNPX geometry to be distortion-free. Then, dose estimation was successfully performed after converting the virtual model into an MCNPX input. A voxelized model was constructed for comparison purposes. The dose estimation functions of the two models were found to be similar, showing a similar amount of computing time by using the mesh tally option with 2e7 histories.

***Keywords**: MCNPX, Dosimetry, Tetrahedron, Phantom*

PS2 (T2.2-0983)

Quantitative Evaluation Methodology for Radiation Dose caused by Transport and Accumulation of Rare-Isotope Beams

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Isotope Separation On-Line (ISOL) system in a Korean heavy-ion accelerator complex called RAON plans to produce rare isotope (RI) beams of high purity and intensity by shooting 70 MeV proton beam from cyclotron driver onto a Uranium Carbide (UCx) target to induce nuclear fission reaction process. The RI beams are transported into SCL3 driver linac for acceleration through pre-separator room and RIB separation and transportation room (RIB-STR). RI beams are important radiation source to be protected as well as prompt neutrons from UCx target on the operation. Because most of the unwanted RIs from UCx are removed and deposited in pre-separator room besides desired isotopes for ISOL system. Therefore, the pre-separator room is considered to have extremely high-level radiation. The RIB-STR has several components that help to prepare the acceleration of RI beams such as RF cooler buncher, EBIS charge breeder, and A/q separator. The RI beams passing through the pre-separator can be either leaked in the beamline or deposited in the several main components in the middle of transportation. In order to safely use RI beams, their radiation risk due to transport and accumulation should be carefully reviewed in the facility. Normally, traditional inventory codes such as FISPACT, CINDER, and D-CHAIN have been employed to calculate the residual dose together with MCNPX and PHITS. However, these codes are not designed to consider the movement of radioactive material (rare isotopes). In addition, the analytical method for residual dose evaluation due to transport and accumulation of RI beams has not been established yet. So, we have modified the traditional inventory codes suitable for residual dose evaluation caused by RI beams transport and accumulation. Then, based on the modified codes, we have developed the analytical method to evaluate the radiation dose due to the use of RI beams and quantitatively calculated the radiation risk in pre-separator room and RIB-STR in ISOL facility. Fig. 1 shows the conceptual idea of RIs transport and accumulation in the route from ISOL bunker to RIB-STR.

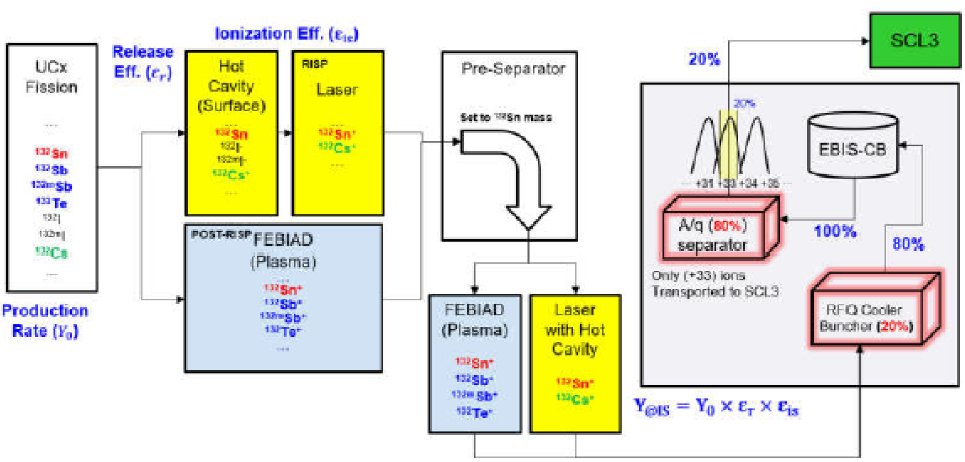

Fig. 1. Transport and accumulation of rare isotope beam in ISOL facility

Keywords: Rare Isotope Beam, Radiation Dose Evaluation, ISOL

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PS2 (T2.2-0986)

Development of 4D Phantom in Moving Postures and Its Application to Assessment of Industrial Radiography Accident Response Scenario

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Recently, a tetrahedral mesh phantom was newly introduced by Yeom et al. [1], which overcame various limitations of voxel phantoms including incapability to define small or thin organs and extremely low deformability. Acknowledging the advantages of mesh phantoms, International Commission on Radiological Protection (ICRP) initiated Task Group 103 with an aim to convert the previous ICRP Publication 110 voxel-type reference phantoms into high-quality mesh format. Currently, Task Group 103 have completed the development of Mesh-type Reference Computational Phantoms (MRCP) for adult male and female, and the publication on MRCPs will be released in this year. The MRCPs are the first reference phantoms that include all the radiosensitive regions required for effective dose calculation, and at the same time, they can be easily deformed into different body shapes or postures. Taking these advantages, Yeom et al. [2] deformed adult MRCPs into five working postures to calculate posture-dependent dose coefficients. The result showed that posture can have a significant effect on organ doses, emphasizing the importance of consideration of postures. However, the posture deformation methodology developed by Yeom et al. [2] involves ~12 hours of computational time and a few days of additional manual modifications, and thus is inappropriate for dose calculations for actual motion which require posture deformation for each of time frames. Therefore, in the present study, fully automatic and rapid posture deformation technique was developed and used to implement 4D phantom in moving postures. In addition, by coupling the developed technique with motion capture system, we developed 4D Monte-Carlo dose calculation program to calculate time-dependent organ absorbed and effective doses by using the 4D phantoms synchronized to the motion capture data. Then, to investigate the feasibility of the application to the assessment of the accident response scenarios, the 4D Monte-Carlo dose calculation program was used to calculate doses in an accident response scenario for industrial radiography source hang-up. During the investigation, current limitations of the program and possible solutions were also verified for further improvement.

***Keywords**: Mesh phantom, 4D Monte-Carlo dose calculation, Motion capture*

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PS2 (T2.2-0996)

Recent advance in computational human phantoms research in USTC: From Population-Averaged Phantoms to Patient-specific Dosimetry using Automatic Organ Segmentation

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Purpose: To present a patient-specific CT organ dose assessment method for radiation risk assessment and procedure/equipment comparison.

Materials and methods: Population-averaged whole-body computational phantoms are the most common research tool for Monte Carlo based organ dose calculations for CT dosimetry. However, in many applications, patient-specific images are available and can be used to derive patient-specific CT organ doses. One barrier has been the lack of automatic CT image segmentation for radiosensitive organs. This paper demonstrates the feasibility of automatic multi-organ segmentation using latest machine learning methods involving novel convolutional neural network (CNN) models, followed by dose calculations using GPU-based ARCHER Monte Carlo code that can compute in real-time. A library of previously developed USTC phantoms is used in this work, as shown in Figure 1.

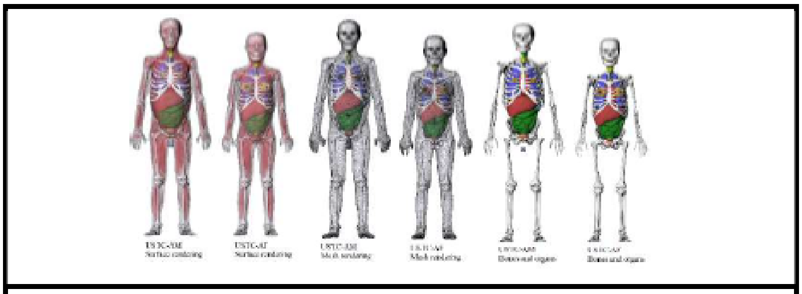

Fig. 1. Previously developed USTC phantoms

We design a novel method using deep convolutional neural network (CNN) to combine features from multiple views and multiple scales for CT image segmentation. The proposed network gradually fuses features from images in different scales with hierarchical receptive fields. Training data include thousands of CT images and their corresponding RT structures finished by doctors. 3D CT volumes are then segmented separately in three views, i.e. axial, sagittal and coronal, and the results are fused together to achieve the final segmentation. The structure of the designed CNN is shown in Fig. 2.

PS2 (T2.2-1063)

Coupling of VRdose and COSSAN for Uncertainty Quantification and Optimisation of Radiological Protection Planning

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Radiation transport simulation is a key tool in modern day radiation protection where accurate dose uptake measurements are achieved by implementing either a Monte Carlo or deterministic algorithm. However, the choice between the two techniques carries an inherent comprise of either efficiency or accuracy. VRdose, developed by IFE in Norway, is an innovative tool designed for the optimization of radiological protection planning. This software implements the deterministic point kernel approach for real-time dose calculations of dynamically changing environments. In the GUI nuclear workers can be modelled undertaking tasks within a radiation environment and hence their dose uptakes estimated such that planning can be made in accordance with ALARP [1]. This study presents a novel code couple in order to extend the optimisation capabilities of VRdose and provide the user with more information regarding the effect of design and planning permutations.

COSSAN is a dedicate uncertainty analysis tool currently being developed at the University of Liverpool; it allows for the integration of 3rd party deterministic software with stochastic numerical sampling algorithms and high-performance computing. This coupling of the deterministic and stochastic allows for analysis of inherent uncertainties which purely deterministic software neglects [2]. The methodology implemented provides the extension of uncertainty quantification whilst retaining the efficiency characteristic of the deterministic technique. The study includes an outline of the computational methodology which includes modification of the VRdose software itself. The VRdose source code, kindly provided by IFE, is executed through a custom-made Java script rather than the GUI and is designed to read and output text files of the relevant parameters. COSSAN then iteratively modifies the parameters of the input files and executes VRdose as a black box in order to produce parametrized dose curves.

This ability to iteratively run through a parameter range rather than a single discrete value, and be automated and efficient, introduces a number of significant advantages which are not currently exploited. Case studies are presented to demonstrate each. The most obvious application is to obtain the possible dose distribution for a complex scenario where parameters are uncertain. On the flipside, the iterative procedure and resultant distribution are used to find the optimum design and planning parameters for minimization of dose uptake, with no need for laborious trial and error. Another feature of COSSAN allows the user to perform sensitivity analysis, when performed on the VRdose-COSSAN couple this function informs the user of the parameters with the most impact on dose uptake and hence aids design and work planning aiming to reduce it. Whilst this project has added functionality which aids the optimisation of work planning and worker protection, the techniques implemented are also applicable to the wider scientific community.

Keywords: Modelling, Optimisation, Protection

ACKNOWLEDGMENTS
I would like to acknowledge the ongoing support of the Institute for Energy Technology (IFE), Norway.

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PS2 (T2.2-1065)

Targeted alpha immunotherapy of CD20+ lymphoma model: Dosimetry estimate of 225Ac-DOTA-rituximab using 64Cu-DOTA-rituximab

Purpose: To evaluate the CD20+ lymphoma using ^{64}Cu-DOTA-rituximab PET and estimate the therapeutic effect the ^{225}Ac-DOTA-rituximab.

Methods: CD20 expression was evaluated in lymphoma cell lines (Jurkat and Raji). DOTA-rituximab was optimally conjugated and chelated by ^{64}Cu. Tumor xenograft models were established in balb/c-nu mice. Animal PET/CT imaging was obtained after tail vein injection with or without pre-dose received 2 mg cold rituximab. Specific binding of tumors was evaluated by organ distribution assay and autoradiography. CD20 expression in tumor tissue was evaluated by immunohistochemistry. The residence time was calculated followed ^{64}Cu-DOTA-rituximab PET/CT acquisition data using the OLINDA/EXM software. The ^{225}Ac-DOTA-rituximab dosimetry analysis was performed the Monte Carlo simulation.

Results: CD20 protein was highly expressed in Raji cells. Specific binding of Raji cells was 90-fold higher than Jurkat cells *(P < 0.0001)*. Immunoreactivity was more 75%. PET/CT imaging with ^{64}Cu-DOTA-rituximab was specifically observed in tumors without pre-dose injection group. Radioactivity of tumor was much higher than other organs and tumor uptake was related to CD20 expression. The predictable human dose for administration of ^{64}Cu-DOTA-rituximab was measured as effective dose (3.20E-02 mSv/MBq). In the tumor region, absorbed dose of ^{64}Cu- and ^{225}Ac-DOTA-rituximab showed that the radiation burden in tumor region of ^{225}Ac-DOTA-rituximab was much higher (76-fold) than ^{64}Cu-DOTA-rituximab ($P < 0.01$).

Conclusion: We evaluated not only the ^{64}Cu-DOTA-rituximab PET/CT imaging in CD20+ lymphoma models but also the alpha-particle immunotherapy with ^{225}Ac-DOTA-rituximab using Monte Carlo simulation. We found that alpha-immunotherapy with rituximab have a high potential to therapeutic benefit in CD20+ lymphoma.

PS2 (T2.2-1071)

Dose evaluation using MCNP6 for μSR facility in RAON

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Rare isotope Accelerator complex for ON-line experiments(RAON), a heavy ion accelerator under construction by Rare Isotope Science Project(RISP) in South Korea, is scheduled to be completed in 2021. One of the RAON's experimental facilities, Muon Spin Relaxation (μSR), is a facility using muons generated by 600 MeV colliding protons into graphite target. The μSR facility consists of a target system, beam dump system, beamline system, and shielding structure, as shown in Figure 1. The area outside the shielding structure is radiation controlled area. The μSR facility radiation protection guidelines limit the amount of radiation in controlled area to be less than 5 μSv/hr considering 50% safety margin.

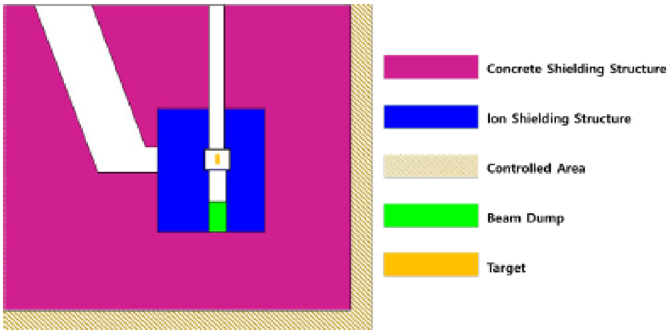

Fig. 1. Structure of μSR facility

In this study, dose evaluation was conducted for the μSR facility operating at 100 kW (125 μA) proton beam power. The code used for dose evaluation is Monte Carlo N-Particle code (MCNP6.1, [1]). The Automated Variance Reduction Generator (ADVATG3.03, [2]) was used to effectively calculate particle transport in MCNP6. The effective dose conversion coefficients were taken from the reference value specified in ICRP publication 116 [3]. Secondary neutron, which has dominant effect on the effective dose, is produced in target and beam dump by protons transmitted from proton beam line. To make efficient calculations, each neutron source term in target and beam dump was calculated. The geometry of the target is azimuthal asymmetry with respect to the beam direction. In contrast, the beam dump geometry is symmetry. Thus, the source term of the target was calculated in all directions, and the beam dump only considered the polar angle. As a result, the effective dose outside the shielding structure was estimated to be less than 5 μSv/hr. In order to evaluate the effective dose in more detail, it is planned to take account of residual radioactivity by calculating concrete activation.

Keywords: RAON, μSR, Dose evaluation, MCNP6

ACKNOWLEDGMENTS
This work was supported by the Rare Isotope Science Project of Institute for Basic Science funded by Ministry of Science and ICT and NRF of Korea (2013M7A1A1075764).

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PS2 (T2.2-1096)

Simultaneous irradiation of multiple ring dosimeters in calibration conditions: influence in the absorbed dose due to the scattered radiation

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Ring dosimeters used for personal dosimetry are calibrated in accredited laboratories using the ISO rod phantom. In principle, the dosimeters should be irradiated in reference conditions, individually. However, the simultaneous irradiation of several dosimeters would reduce time and costs. Nevertheless, the radiation scattered by a dosimeter could influence the overall absorbed dose to another dosimeter. The aim of this study was to determine, through Monte Carlo (MC) simulations with the code PENELOPE, the additional uncertainty of the calibration factor of $H_p(0.07)$ if three or five ring dosimeters are irradiated simultaneously, for different photon beam qualities.

The use of MC simulations in this kind of studies is justified by their advantage over experimental methods in the reduction of the uncertainty of the studied magnitude. As an input for these simulations, a mathematical model of the DXT-RAD 707H-2 (Thermo Scientific) ring dosimeter was developed. Then, three different configurations were simulated: one single dosimeter in the centre of the phantom (reference simulation); three dosimeters equidistant 2.10 cm and five dosimeters equidistant 1.05 cm. The simulations for each configuration were repeated for three different parallel photon beam qualities: N-30, N-80 and N-300. The absorbed dose to each dosimeter in the simulations was compared with the one by the reference simulation. The simulated results were validated with irradiations at the Secondary Standard Dosimetry Laboratory (SSDL) of the Spanish National Dosimetry Centre (CND), for the configuration with three dosimeters.

For all beam qualities, the absorbed dose to each dosimeter is compatible with each other and with the absorbed dose to the dosimeter simulated in reference conditions, both with ~95% of coverage probability. The geometry and the simulation for the configuration with three dosimeters were verified with experimental data from an irradiation performed at SSDL, finding consistent results with the same coverage probability (~95%), with the MC simulations having a lower uncertainty (~4%, k=2) than experimental data (~10%).

In conclusion, there exists no significant mutual influence in the absorbed dose by each dosimeter and the ring dosimeters can be simultaneously irradiated, at least in groups of five equidistant 1.05 cm, with a negligible impact in the estimation of the uncertainty of $H_p(0.07)$.

***Keywords**: Personal dosimetry, ring dosimeters, calibration, Monte Carlo simulation.*

PS2 (T2.2-1128)

Estimation of the radiation dose absorbed by a patient in thyroid gland and the lens during the taking of an oral radiography

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The radiation dose that is received during a dental x-ray is low compared to what is received during other radiological procedures, however, of every four radiographies that are done, one is dental. On the other hand, a large proportion of dental radiographies are done to children and adolescents, people for whom the risk of exposure to X-rays is twice as high. Another thing that worries a lot during the taking of a dental radiography, is the exposure of radiosensitive organs such as the thyroid gland, the brain, the salivary glands and the lens. In addition to this, in dental radiology, many patients may be undergoing unnecessarily high radiation doses, due to obsolete or decalibrated equipment, inadequate position and exposure techniques and poor image processing. In this regard, the National Council for Radiation Protection and Measurement (NCRP) indicates that exposure during radiological examinations has increased considerably in different organs because these examinations are not being performed taking into account the clinical needs of patients

The determination of the radiation dose in diagnostic radiology is a critical factor in optimizing the radiological protection of health professionals, the patient and the public [1]. In diagnostic radiology, the main interest is to estimate the response of biological tissues to low doses of radiation, as is the case of radiation doses in dental radiology. It is not possible to do this directly and some studies have estimated the doses of radiation absorbed in the most radiosensitive tissues using computer programs that simulate the interactions of ionizing radiation with these organs.

One of the objectives of this investigation was to estimate the radiation dose absorbed by a patient in the lens and thyroid gland with exposure and positioning protocols applied to panoramic and intraoral dental radiography equipment, the estimation was made by implementing a code based on Monte Carlo in the Geant4 computing tool. The influence that the type of collimator and the choice of technique had on the dose of absorbed radiation was determined. For the simulation has been implemented a head and neck computational anthropomorphic phantom with thyroid and crystalline delimitation, the material associated with the phantom was liquid water. Table 1. shows the doses absorbed by the lens and thyroid gland of a patient.

Table 1. Dose absorbed ($x10^{-5}$ pGy), by incident photon, in the thyroid gland and in the lens of a patient during a dental radiography taken with intraoral equipment (left) and a panoramic (right).

Kilo voltage	Thyroid gland	Lens
50kVp	$4,6 \pm 0,1$	$4,8 \pm 0,3$
60kVp	$6,8 \pm 0,2$	$6,8 \pm 0,3$
70kVp	$7,9 \pm 0,2$	$6,8 \pm 0,3$

Kilo voltage	Thyroid gland	Lens
50kVp	$7,6 \pm 0,2$	$8,5 \pm 0,3$
60kVp	$9,4 \pm 0,2$	$12,0 \pm 0,4$
70kVp	$10,5 \pm 0,2$	$12,6 \pm 0,4$

Keywords: dental radiography, dose, radiation protection.

ACKNOWLEDGMENTS
We thank ECCI University for the technical and financial support offered to this project.

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PS2 (T2.2-1155)

The Study of Atmospheric Dispersion Factors for Radiological Analysis at YONGGWANG(HANBIT) Site in KOREA

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In order to estimation the safety of NPP(Nuclear Power Plants), air pollution model have been studied by the IAEA, NRC, and EPA. The purpose of this study is to improve a more efficient methodology of explaining the atmosphere behavior characteristic and the radiation dose characteristic depended on the atmospheric stability. Generally, Lagrangian, Eulerian or Gaussian models have been used to explain the air pollution behavior in NPPs. These models include various methods to determine the stability of the atmosphere. Also, the models are very important to estimate the radiation dose due to the proportional increase by atmospheric dispersion factor. In this study, various models and simulations are reviewed. And the respective methodologies are applied to YONGGWANG site to identify benefits. The key parameters of major atmospheric stability by classification methods are evaluated. Using the meteorological data obtained by the data acquisition system of the YONGGWANG site, the various atmospheric stability, dispersion factors, and correlation factors of radiation dose are totally confirmed and estimated. The Pasquill's classification method is the most widely used on the basic methodology and it is good agreement with a steady state and rapid response condition under timely stability by heating or cooling caused by daily insolation or nocturnal surface radiation. In this study, from a comparison of modeling calculation results, we find that horizontal and vertical standard deviations of the wind fluctuation method tend to analysis night-time stable conditions such as unstable conditions. In this study, the classification matrix methodology of Vogt's vertical temperature difference and wind speed is changed for practical use at the YONGGWANG site. From this study, it is also confirmed that the methods using the bulk Richardson number and Vogt's modified graphs are very useful in classifying atmospheric category. Using RG 1.23, the modified tables for delta T and wind speed U and the Richardson method were made by calculating the reasonable joint frequency distribution from Pasquil's method and other existing results. From the obtained results, the correlation coefficient (r) was equal to 0.931. The results of radiation dose are proportional to atmospheric dispersion factors. From study results, the most optimization among the various methodologies is made by comparison with each other. From this study, the best estimation methodology is the case of the delta T and wind speed. The methodology is more than 0.9 in correlation factors under the low wind speed condition and the high wind speed condition. In this methodology, the correlation values between the atmospheric dispersion factor and the radiation dose is more than 0.91. Each method is compared and the key parameters are checked in the case of wind speed, stability, wind-direction. Key parameters are the delta T and the wind speed. In this study, the various comparison results of each model are introduced.

Keywords: Atmospheric dispersion factors , Stability, Radiation Dose, Correlation, ICRP, FGR, NRC

ACKNOWLEDGMENTS

This study has been partly supported by the safety issue program and the program of Periodic Safety Review of YONGGWANG site in KOREA.

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PS2 (T2.2-1156)

The Study of Iodine Spiking Phenomena on Pressurized Water Reactor in KOREA

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When the fuel rod of Nuclear Power Plant defects, the cladding materials are no longer the protector between the gap atmosphere of the internal fuel material and the primary coolant. In this time, a leak path as the crack then appears so that the RCS coolant can enter the gap between the fuel rod and the clad. The fission products in the gap can escape and enter into the reactor coolant system (RCS). A various insight has provided a better understanding of the physical processes of activity release during the reactor's operating at steady-state power since 1970. In full-power operating state, only a small fraction of the fission-product iodine is released into the RCS.

Most of the released iodine is present as a liquid soluble deposit state on the fuel surface or the inner surface of cladding. When the temperature of the gap between the pellet and the cladding drops, the clad temperature can be dropped below the coolant saturation temperature during reactor shutdown. The water entered into the rod is remained as the liquid phase and is deposited on the cladding surface. In this study, the behavior of iodine is estimated and explained in detail. The estimation method includes the sensitivity results, the correlation with NPP's thermal power, iodine diffusion mechanism, and so on. From this result, the conservatism of NRC's methodology is appeared. The various results of checking the detailed iodine-behavior is introduced. Finally, the radiation dose is calculated and the calculated value is proportional to the spiking-factor.

Keywords: Iodine-spike, Spiking phenomena, Radiation Dose, NRC

ACKNOWLEDGMENTS

This study has been supported by the safety issue program and the program of Periodic Safety Review in KOREA.

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PS2 (T2.2-1180)

Theoretical simulation and experimental study on material activation for the key components of NBI

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The neutron source intensity in the plasma center is on the order of 1E+20n/s by DT fusion reaction, during the China Fusion Engineering Test Reactor (CFETR) fusion power operating with 1.5GW. Fusion neutrons stream easily into the interior of the neutral beam (NB) modules through the wide ports and the drift ducts. Since there are no obstacles in the neutral beam path, a very high neutron flux penetrated the beamline and activate the NB module. The shutdown dose rate caused by activation will impact for the neutral beam injection's maintenance man. Also, the material stability will be effected by the decay heat released from the material activation. So it is important to calculate the material activation for the neutral beam injection (NBI).In this paper, the Monte Carlo method and the two-dimensional RZ geometry model are used to analyze the neutron fluxes and the materials activation for NBI. The results demonstrate that the induced radioactivity of the Calorimeter (CAL), the Residual Ion Dump (RID) and the support components of the CAL and RID are Cu and SS316, during the CFETR operation. And then, the activation samples were designed in order to observe the activation results experimentally. Then, based on the HINEG device, the copper sample is irradiated 20 minutes by DT neutron source. Next, we measure the gamma-ray spectra of the samples using the high purity germanium detector. Theoretical simulation and experimental measurement results are shown in Fig. 1 and Fig. 2.

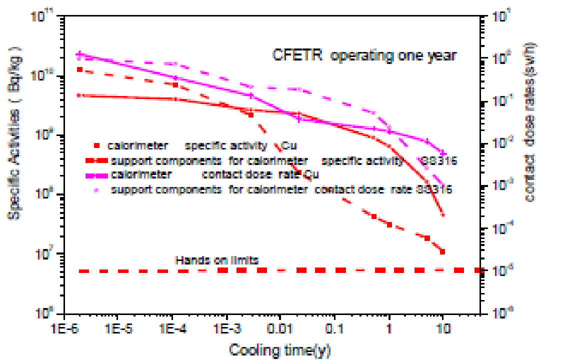

Fig.1 The specific activity and contact dose rates

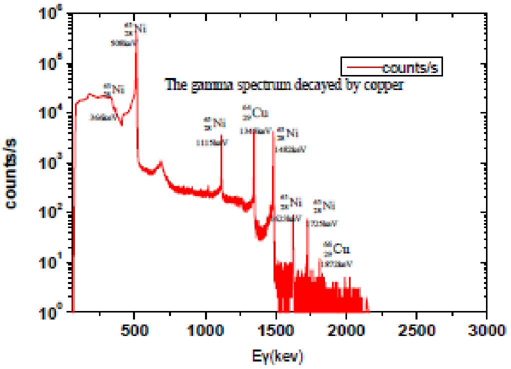

Fig.2 The gamma spectrum decayed by copper.

Keywords: NBI, Material activation, Montle Carlo method

ACKNOWLEDGMENTS

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PS2 (T2.2-1216)

Individual dose assessment in ^{99m}Tc-GSA hepatic scintigraphy using a nine-compartment biokinetic model

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Technetium-99m-galactosyl human serum albumin (^{99m}Tc-GSA) is a molecular imaging agent used for evaluation of hepatic function. Sixty-two examinations of ^{99m}Tc-GSA hepatic scintigraphy have been performed from December 2018 to November 2019 at QST Hospital in Japan before and after heavy-ion therapy for liver tumors. In order to enable to estimate individual variation on absorbed doses for each patient, a method of internal dose assessment for the organs and tissues in ^{99m}Tc-GSA scintigraphy was developed using a nine-compartment biokinetic model (Fig. 1) taking advantage of the existing models[1, 2, 3)]. The developed method is also expected to give a better understanding for metabolic changes along with progress of diseases. According to the model shown in Fig. 1, the transfer rates between compartments regarding the excretion pathways (i.e., from bladder to urine, from small intestine to faeces) were determined to be constant according to the ICRP recommendations[2, 3)]. The other transfer rates were optimized for each examination by comparing with the retention functions of liver and heart clinically observed from individual dynamic scanning with the gamma camera (Siemens E.cam Signature Series) for the trunk region in a duration of 31.5 minutes (30 seconds/frame) after intravenous injection of ^{99m}Tc-GSA. For this comparison between clinical observation and model prediction, the ratios of blood volume in liver and heart were assumed to be 10% and 9% of the total blood volume, respectively[4)]. The simultaneous differential equations for the activity in each compartment were numerically analyzed using a commercial software, EQUATRAN-G, and obtained the total amount of disintegrations taken place in each compartment for 2.5 days corresponding to about 10 physical half-lives of ^{99m}Tc. Absorbed doses of the organs and tissues were calculated using the internal dose calculation software, IDAC-Dose 2.1[5)]. As a result, an average dose per unit injected activity (mGy/MBq) for the liver among the examinations was calculated to be higher than doses for the other organs and tissues and was about 20% lower than that estimated for healthy male subjects (i.e., 0.054 mGy/MBq)[6)]. A further extension of the model regarding extrahepatic biliary system remains to be discussed for improving the doses assessed in this study.

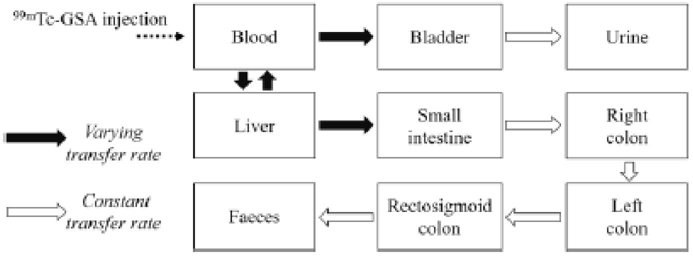

Fig. 1 Nine-compartment biokinetic model

Keywords: ^{99m}Tc-GSA, internal dose assessment, biokinetic model

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PS2 (T2.2-1228)

Monte Carlo simulation for optimization of thyroid measurements

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Ionizing radiations could cause external or internal exposure to the human body. Usually the internal exposure is determined in two steps. In the first step the actual activity present in the body is determined by direct or indirect monitoring methods. By direct measurements activity in the whole or part of the human body can be determined (*in-vivo*). In the second step the intake value and the associated committed dose can be estimated on the basis of measured data considering necessary assumptions on exposure conditions (time and route of intake, chemical form etc.). In case of the *in-vivo* measurement the low activity of the human body, the non-standard geometry and the limited measurement times considerably increase the measurement uncertainty. Thyroid measurements are also influenced by various parameters e.g. the shape of thyroid, position of the thyroid inside the human body, the detector distance from the body surface, the distribution of the activity within the organ and the position of the detector during calibration with a thyroid phantom. After efficiency calibration of the thyroid counter with physical phantoms, efficiency calibration was performed by numerical methods to optimize the measurements. To account for uncertainties introduced by these factors in the estimation of different radioactive isotopes of iodine numerical simulations based on Monte Carlo photon transport techniques were performed, detector response and corresponding detection efficiencies were calculated. During our studies uncertainties due to different detector sizes (±1 cm), differences in the distance of the thyroid from neck surface (0.6-2.2 cm), different shape of thyroid (e.g. thyroid of child, teenager and adult) and the detector distance from the body surface (4-12 cm) were analysed. All these factors affect the accuracy of dose estimation as well. The computed uncertainties due to various parameters should be taken into account while estimating the activity of iodine isotopes in the thyroid.

***Keywords**: Thyroid, Monte-Carlo simulation, In-vivo measurement*

PS2 (T2.3-0260)

Examples of Applications using Continuous Sequential Repeating Quantitative Gamma Spectral Acquisition and Analysis

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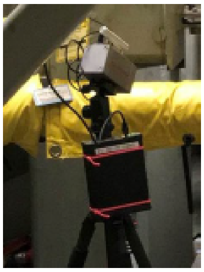
Many nuclear facilities have on-line instruments that continuously record a radiation signal, but these generally record count rate, or dose rate. These instruments are quite helpful to record the presence or absence of radiation changes. However, if after a significant change in the instrument reading the actual radionuclides involved and their activities must be determined; this requires extracting a representative sample, and after some delay, a laboratory analysis.

The Data Analyst [DA] is a device that connects to CZT, Scintillation or HPGe gamma detectors and their respective Multi Channel Analyzers. The DA is an autonomous device that upon startup automatically starts acquiring a repeating sequence of gamma spectra for a user-defined time. After the time expires, the next spectra is started with no lost time, and the previous spectrum is analyzed and the result made available. Alarms can be set if the activity is above the user's threshold. Acquisitions can run continuously or be triggered by an external signal.

The following are a few of the applications that have been served using the DA and either a CZT, NaI or HPGe gamma spectrometer:



- Primary coolant at Nuclear Power Plant; EPRI has partially funded several demonstration measurements using a shielded collimated CZT detector aimed at a pipe. These have been performed at 4 different reactors. Generally these covered an outage cycle, but one application will last the entire fuel cycle: >2y.
- Primary coolant at Nuclear Power Plant: EPRI is partially funding the construction of an HPGe system that will measure an extracted primary coolant sample. The device will measure the fresh coolant continuously most of the time, but periodically will extract and decay a sample and then count it.
- Stack Gas Monitor: Three units have been delivered to various reactor sites in Europe, USA, and Australia. These all have a shielded HPGe detector and a large Marinelli beaker for the gas. They are designed to measure a very wide [8 decade] dynamic range of gas concentration.
- Fuel Rod Scanner: This has a heavily collimated HPGe detector viewing a 0.5mm wide section of the fuel rod. Here the triggered mode of the DA is used to acquire a single spectrum, then the fuel rod is moved a bit, and the sequence repeated.
- Pipe Monitor: A waste processing facility is using a collimated NaI unit to evaluate key nuclides. If their activity is too high, a signal is sent to divert that material for reprocessing.
- Robotic measurements of ground and floor activity: A robot is programmed to autonomously drive a predetermined pathway. Suspended from the robot are two separate NaI spectrometers and two DA modules. Nuclide analyses are sent to the robot every 3 seconds from each detector.

Keywords: Gamma, Mapping, Spectrometry

PS2 (T2.3-0316)

Quality assurance of the radioactivity measurement using gamma spectrometric method

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One of the technical requirements of ISO / IEC 17025: 2017 is that the laboratories that test and calibrate verify that they can properly carry out the methods before using them, ensuring that the required performance can be achieved. Among the tests carried out by the Environmental Radiological Surveillance Laboratory of the Center for Radiation Protection and Hygiene of Cuba, is the determination of radionuclides by high resolution spectrometry. This test has been accredited before the National Accreditation Body of Cuba recently, in accordance with ISO / IEC 17025: 2017. In compliance with the requirements of this standard, in relation to the selection of test methods, the laboratory uses an internationally standardized test method for its spectrometric gamma determinations. For this reason it was necessary to confirm the correct application of the selected method before being incorporated into the scientific-technological services offered by the institution. To this end, different method checks were carried out, using certified reference materials for a varied gamma of matrices analyzed in the laboratory and for the different measurement geometries used. The results of the measurements obtained were statistically processed to verify their accuracy and veracity. The laboratory also periodically participates in interlaboratory comparison exercises, which include the analysis of radionuclides by gamma spectrometry, as another form of confirmation of the accuracy of the results reported by the laboratory. This paper presents both the design of the intralaboratory study carried out, as well as the results thereof. From the results obtained that the study carried out, as well as in the interlaboratory comparison exercises in which it has participated, it can be concluded that the method used in the laboratory is properly executed and is under control.

***Keywords**: gamma spectrometry, precision, trueness*

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PS2 (T2.3-0319)

Continuous radioactive liquid effluent monitor base on NaI detector

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Many nuclear facilities discharge low levels radioactive liquid to environmental such as rivers, lakes, etc., which would become domestic water and even drinking water. Reasonable Continuous monitoring of these liquid radioactive effluent is a prerequisite for control. It was necessary to develop a better detection limitation of continuous liquid radioactive effluent monitor device, because most these types monitor only had the detection limitation range from 10000 to 50000Bq/m3 with none radionuclide identification capacity and were difficult to meet radiation protection and environmental protection needs. In this paper, a radionuclide identification continuous liquid radioactive effluent monitor based on NaI detector is developed for the measurement of the key gamma radionuclide such as co-60 and cs-137 in nuclear facilities. The monitor had been tested for more than 550 hours, results shows it is stable and reliable, has the ability of radionuclide identification, detection limitation was improved.

KEYWORDS: Liquid radioactive effluent, monitor, multi-channel, detection, discharge

PS2 (T2.3-0359)

Development of a Rotational Modulation Collimator Imaging System with an Enhanced Field of View

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A Rotational modulation collimator (RMC) imaging system has been investigated for the radiation safety and homeland security application [1-4]. This technique has the advantage of eliminating the need for position-sensitive radiation detectors, offering the possibility of reducing the system complexity and cost. Using the intensity of incident particles at each rotation angle of the mask, one can obtain a modulation pattern of detector counts in the angular domain, which is used to reconstruct a radiation image (see Fig. 1). Thus, it can turn a non-position-sensitive radiation detector into a radiation imaging system. However, since the basic structure of RMC is cylindrical geometry, it has a limited field of view ~20° in the cross-sectional plane view. Therefore, as an enhanced design for the RMC system, we proposed a hemispherical collimator design which can extend the FOV approximately up to ~2π [4]. In the present study, measurement experiments were conducted with gamma-ray sources for testing the hemispherical RMC (H-RMC) imaging capability, and reconstructed images were evaluated with the quantitative evaluation factors. Measured modulation patterns showed good reproducibility, and the actual source position was estimated correctly as a point having the largest maximum likelihood estimation value in the reconstructed images. In this study, we demonstrate the imaging capability of the proposed system, which can describe the promising prospects for applying the H-RMC system to radiation detection and monitoring application. The H-RMC system is expected to utilize an effective tool for imaging the spatial distribution of radioactive materials.

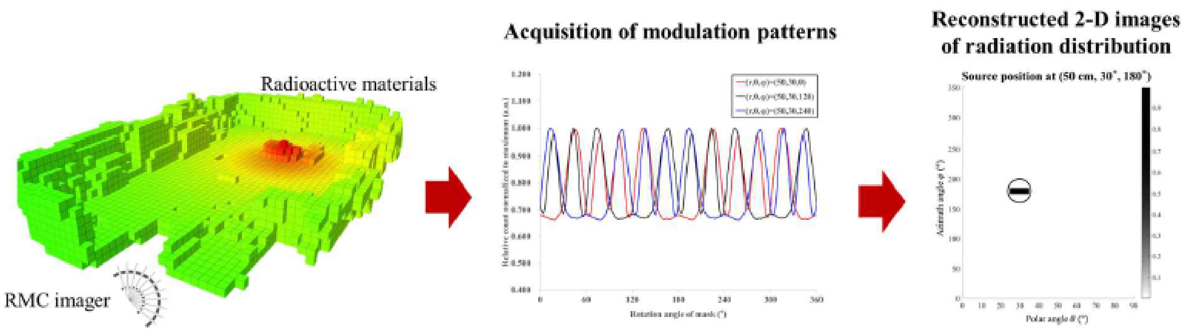

Fig. 1. Overview of the RMC imaging technique.

Keywords: *Radiation detection and monitoring, Rotational modulation collimator, Radiation imaging system*

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PS2 (T2.3-0417)

IPCM12 Radon Enhancements

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Thermo Fisher Scientific

False positive nuisance alarms caused by radon have presented a health physics challenge since the introduction of Alpha/Beta Personnel Contamination monitors. Such nuisance alarms result in reduced productivity, extra health physicist involvement/investigation, and lower user confidence in the instrument.

The improvements recently implemented into the IPCM12 result in a significantly better, highly accurate, identification of system counts due to radon that reliably differentiate system counts due to actual alpha or beta contamination. This leads to increased worker productivity, increased worker confidence, and reductions in operating costs while maintaining proper radiation contamination controls.

PS2 (T2.3-0715)

Applications of dosimetric measurements to Quality studies and optimization of operating conditions at a linear accelerator

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Radiation therapy based on the use of an accelerator plays an important role in the treatment of malignant tumour. The quality control for linear accelerator is one of the keys to ensure the correct and safe implementation of accurate radiotherapy.
The National Center of Oncology in Nouakchott is equipped with a linear accelerator which provides two energies in photon regime 6 MV and 18 MV.
A set of dosimetric measures were performed within this accelerator. Percentage Dose Depth (PDD) measurements and comparisons to calculations using the Treatment Planning Systems (TPS) provided by the IAEA were performed to study the quality of the accelerator. These same measures were used to optimize the operating conditions of the accelerator. This optimization consists to determine the ideal dimension of the field size for each energy and the best dimension of the ionization chamber used as detector.
The PDD measures have been determined for 6MV and 18MV beam photon energy, for four different dimensions of the field size.
The all measured results were comparable for all chosen treatment field dimensions to those calculated by TPS.

Keywords: *Radiotherapy, Linear accelerator, dosimetric measures*

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PS2 (T2.3-0760)

The "GIRAFFE": a dedicated tool for gamma and neutron dose rates measurements on packages loaded with spent fuel assemblies

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During the transport of a cask loaded with spent fuel assemblies from an EDF nuclear power plant, the thresholds on the maximum gamma and neutron dose rates around the package and the conveyance imposed by the Safe Transport of Radioactive Material regulations must be complied with. The gamma and neutron dose rates are evaluated at 70 points around the cask at 2.60 m from the ground, at the contact and at 1 m from the cask and finally at 2 m from the vehicle by means of a neutron monitor and a gamma meter.
Currently, the neutron dose rate is evaluated by using the PNM200 (MIRION), a 6.5 kg neutron monitor handled manually by the worker. It is evident that the worker is subjected to a non-negligible physical stress when realizing this kind of measurements: the monitor must be kept still at 70 points for 100 s each, at 2.6 m in height with thus the fall risk present. Moreover, the worker is exposed to the gamma and neutron radiations for the whole term.

In the joint effort of enhancing the safety and reducing the dosimetry of the workers, without compromising the quality of the measurements, EDF has solicited bids for the conception and the development of a tool allowing the workers to realize safely quality measurements.

This tool, named the "giraffe", is a system mounted on a trolley equipped with four all-terrain wheels, disposing of a case for the gamma meter and of a grip for the neutron one. The two devices can be plugged on a tablet for the remote reading and storing of the measurements. The "giraffe" employs an adjustable telescopic mast which allows the placing and maintaining of the meters at the right position. Finally, four fold-up stabilizing feet make the system secure.

This contribution reports on this tool, the "giraffe", and on the tests carried out to approve the prototype. Also, a tentative extrapolation of the dosimetry gain will be given.

Figure 1. A neutron dose rate measurement at the contact of the cask using the "giraffe" with the PNM200.

Keywords: *neutron, fuel, dosimetry*

PS2 (T2.3-0764)

Comparison of experimental and MCNP6 results for a LUPIN-II neutron REM-meter at the UPM neutronics hall

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LUPIN-II[1] is a recently developed wide-range neutron REM-meter, designed for the control of the ambient dose equivalent H*(10) due to neutrons fields generated in high-energy pulsed facilities, like high-energy particle accelerators. Hence the characterization of this device could be very useful for different applications.

The aim of this work was to develop and test a realistic model of the LUPIN-II REM-meter for the Monte Carlo code MCNP6[2], which could be used for different application scenarios, like those in hadron therapy facilities. A detailed realistic model of the neutron area monitor was designed including the cylindrical high density polyethylene body, the BF_3 detector, and the Pb and Cd inserts. This model was combined with the MCNP6 realistic models of the neutronics hall at UPM[2], including a 111 GBq ^{241}AmBe neutron source, an irradiation bench and the whole large room. Some irradiation tests were performed with the ^{241}AmBe source at different distances and the simulations with MCNP6 code were compared to the experimental results. This comparison has allowed establishing a mathematical relationship, semiempirical model, between the numerical and experimental results at different source-to-detector distances in the irradiation bench, updating the values of H*(10). These results have been also compared with those obtained with a conventional Berthold LB-6411 neutron REM-meter in order to benchmark experimental results with both conventional and wide-range REM-meters. The obtained results show a good agreement between the calculated and measured values of H*(10), with deviations between 3% and 9%. However, for the ^{241}AmBe neutron fields of the UPM neutronics hall LUPIN-II tends to overestimate H*(10) rates between 15% and 20%.

***Keywords**: REM-meter, LUPIN-II, MCNP6*

ACKNOWLEDGMENTS

This work has developed under the industrial Doctorate Program IND2017/AMB-7797, funded by the Madrid Autonomous Region (CM).

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PS2 (T2.3-0782)

Development of in situ calibration system for environmental monitoring instruments

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More than 3000 area dosimeters for environmental monitoring are installed in Japan. Most of them are strongly fixed, and it is difficult to remove them for calibrations. There are some methods for in situ calibration for monitoring devices, but there are some problems. One of them is influence of scattering photons from other instruments around a monitor. And another problem is high background at limited area. There are a few area, but the background dose rate is above 1 micro Sv/h. Such a condition, measurement uncertainties are increase for the calibration. In this study, new irradiation system is developed for in situ calibration. The irradiation is done with a collimator to reduce the influence of scattering photons, and this system is relatively lightweight to be portable. And also shielding material is designed to reduce a background radiation for more precise calibration.

The calibration test was done at high background area for one portable area monitor and two fixed area monitors. The portable area monitor had been calibrated at low background area by another calibration method using the reference cavity chamber. Our calibration result agreed very well with previous calibration result.

PS2 (T2.3-0792)

Determination of Correction Factor of Self-Absorption for Pb-210 in Environment Samples using Two Experimental Methods

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This study aimed to determine the self-absorption correction factors for determination of 210 Pb in environmental samples using Gamma Spectroscopy HPGe. Two techniques were used, namely the reference of Cutshall and the standard additions method or spike method , where known amount of 210 Pb is added to the sample and measured twice before and after the additions; Taking care to obtain a homogeneous sample after drying of the samples studied. The first method is based on measuring the gamma line of 210 Pb emitted from a standard source placed above the sample to determine the correction factor due to attenuation.
The results showed possibility to obtain reference curves for the self-absorption correction factors with the density of the samples for each type and have a moderate positive correlation coefficient, provided that the samples from the same environment and have the same geometric shape. On the other hand, it is difficult to obtain reference curves that describes the variations of self-absorption correction factors with the chemical composition of the samples due to the wide spectrum of compounds and elements constituent of environmental samples.

Keywords: *Environmental Samples , Lead210 , HPGe, Gamma Spectroscopy , Cutshall's method , Spike method, Radiation Measurement*

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PS2 (T2.3-0814)

Monitoring and Analysis of Dose Rate and Gaseous Effluent Released from Radioactive Waste Installation in Indonesia

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Dose rate and gaseous effluent released monitoring system (Radmon) that real-time monitoring continuously measures gamma dose rate on 7 (seven) process room of Radioactive Waste Installation (RWI) in Indonesia. Gaseous effluent released monitor is used for RWI stack. Radmon is an important tool to present dose rate and gaseous effluent released information to the public or authorities for radiation protection during normal operation and radiological accidents. We have developed such a system that consist of 7 (seven) NaI(Tl) based device for monitoring the dose rate and gaseous effluent released monitoring system of RWI. It has operated since 2011. In this study, the analysis and the description of measured data of Radmon are presented. The analysis dose rate data for the last 5 (five) years shows that the average dose rate level were between 0.12-3.73μSv/h and The analysis gaseous effluent contamination released data for the last 1 (one) years shows gaseous effluent contamination released for alpha 0.002 Bq/m^3 and beta 0.097 Bq/m^3 which are similar with background radiation level and Radon (222Rn) 2.452 Bq/m^3, Thoron (Tn or 220 Rn) 0.022 Bq/m^3 which are lower than threshold. This result indicates that the system is good situation in normal condition and effective for a radiation early warning system for radiological emergency case.

Keywords: *Dose rate, Effluent release, RWI, Real-time Monitoring, Radiation Early Warning System, NaI(Tl)*

PS2 (T2.3-0846)

Detector Efficiency and Contamination Detection in Nuclear Medicine Services

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The aim of this work was to detect, through controlled contamination of radioactive material, the counts for a pre-established activity using detectors of different characteristics to establish a contamination assessment pipeline and to study the best approaches to reduce contamination and effectively optimize decontamination methods. Four contamination detectors were used: three Geiger-Müller pancake probe detectors (two internationally branded and one made in Brazil), and one spectral detector capable of identifying radioactive elements. The following radionuclides were used: 510 µCi ^{99m}Tc, 550 µCi ^{131}I, 500 µCi ^{67}Ga, 520 µCi ^{68}Ga and 530 µCi ^{18}F. The measurements were taken on a lined bench, where contamination was carried out on a 1 cm^2 cotton area with the elements individually scattered. Measurements were taken from 10 cm to 30 cm distance between the detector and contaminant due to constant scale overflow limitation at 10 cm distance. The Table 1 presents the achieved results.

Table 1. Detector efficiency counting measurements

	Detector 1		Detector 2		Detector 3		Detector 4	
	AVERAGE KCPM	SD	AVERAGE KCPM	SD	AVERAGE KCPM	SD	AVERAGE KCPM	SD
^{18}F	15.336 ± 2.788		*		*		*	
^{99m}Tc	13.274 ± 460		10.260 ± 924		13.152 ± 485		12.789 ± 654	
^{67}Ga	3.946 ± 396		2.923 ± 225		2.956 ± 499		3.665 ± 646	
^{68}Ga	8.757 ± 382		*		*		*	
^{131}I	7.006 ± 1.199		5.679 ± 101		6.412 ± 984		5.833 ± 974	

Legend: SD: standard deviation, F: Fluorine, Tc: Technetium, Ga: Galium, I: Iodine, *: over range

The spectral detector was the one with the best measurement efficiency for the positron emitting elements and the other radionuclides. The other detectors proved to be efficient for specific radionuclides only and were not as efficient at close distances. These preliminary results show that an efficient form could be designed to record what is to be considered as an effective contamination. This is the next intended step in the research.

Keywords: *Detectors, Contamination, Optimization.*

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PS2 (T2.3-0849)

Tritium contamination of wood sample and recovery test in not burning but drying method

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When tritiated water (HTO) is electrolyzed by proton exchange membrane (PEM) cell, gaseous tritium (HT) is generated. And wood samples contaminated by tritiated water and gaseous tritium was pretreated by not burning but drying using air and oxygen gas. When wood sample is dipped in tritiated water, wood sample can be contaminated by HTO molecules. Surface of wood sample can be contaminated by HTO molecules. However, HTO molecules as liquid phase cannot be sunk to inside of wood sample coated by varnish. When there is wood sample in the exit of electrolysis system, wood sample contacts with gaseous tritium and is contaminated by isotope exchange between vapor water molecule in surface of wood sample and gaseous tritium [1]. Also, because wood sample is combusted and becomes ashes in high temperature combustion higher than 450 °C, tritium recovery almost shows 99 %. However, in case of combustion on temperature lower than 450 °C, tritium recovery can be different as phase of tritium in wood sample. Figure 1 shows tritium contamination of wood sample according to phase.

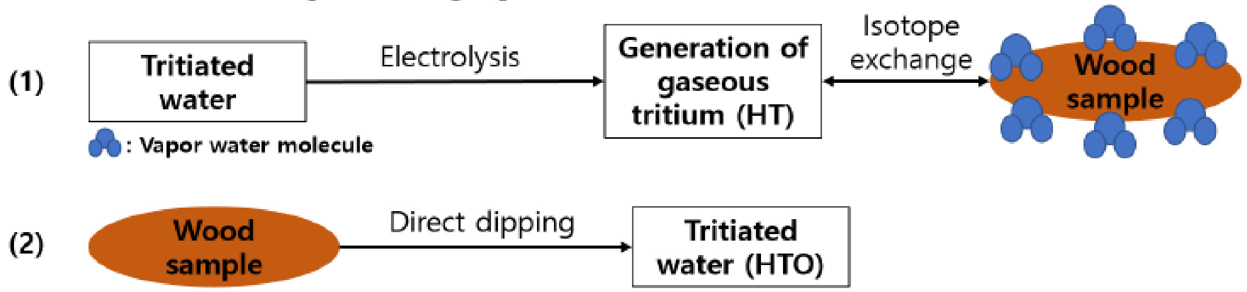

Figure 1. Schematic diagram about tritium contamination of wood sample

Table 1 shows recoveries when wood sample with the radioactivity of about 40 Bq was combusted for 2 hours in various temperature condition. Results shows high recovery when wood sample is combusted in temperature higher than 450 °C. However, recoveries on low temperature condition of 150 °C were 17.35 %, and 21.82 %, respectively. Changing temperature, and combustion time finely, optimal temperature and combustion time with the recovery as high as recovery on burning temperature of wood sample will be found in the future. And then, without making ashes of wood sample, high recovery can be secure and wood sample can be reused.

Table 1. Radioactivity of wood sample at the various temperature

Contamination method	temperature (°C)	Recovery (%)
Electrolysis	150	17.35
	300	46.12
	500	97.79
Dipping in tritiated water	150	21.82
	300	38.88
	500	98.89

Keywords: wood sample, combustion treatment, tritium recovery, tritiated water, gaseous tritium

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PS2 (T2.3-0894)

In-situ measurements of radon exhalation rate from walls: a proposal for an innovative apparatus

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Knowledge of the several mechanism of radon entry into buildings has considerably increased through the years. Among mechanism contributing to radon accumulation into indoor air, diffusion (i.e., driven by concentration gradient) and advection (driven by a pressure gradient) through the building materials of the walls (in the following, "wall" will refer to all indoor masonry surfaces other than floor) both could play an import role. UNSCEAR, in quantifying the different contributions, reports that 18%[1] of total radon entry can be attributed to diffusion from walls by considering distinct house models[2]. Under some circumstances (i.e., tunnels or cavities in building structure getting in communication rooms walls with foundations and soil beneath the building), the advective contribution, often neglected when dealing with walls, can be significant too and even predominant.

Being the radon entry predictive model often too complex and unreliable to apply, in-situ measurements of radon exhalation rate from walls represent the best way to estimate the "walls contribution" to indoor radon concentration. The International Standards Organization, while discussing measurement of radioactivity in the environment, refers to accumulation methods as the unique solution to estimate exhalation from surfaces[3]. Those techniques are based on radon exhalated increasingly accumulating in a container of known dimensions. Due to the difficulty to quantify the influence of environmental parameters and phenomena, this method is declared as only estimative[3].

An innovative apparatus for measuring radon exhalation rate from walls surfaces has been conceived. The design of the system moves from the study of the main influencing factors affecting both the accumulation phenomenon (i.e. air tightness, back diffusion and variation in conditions in and outside the accumulation container) and the radon activity concentration measurements (thoron interference for continuous radon monitors). The main characteristics and features of this apparatus is descripted in this poster.

***Keywords**: radon exhalation rate, in-situ measurements, wall surface*

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Poster Session T2

PS2 (T2.3-0914)

Alpha background study of ^{3}He-filled proportional counter for low-rate neutron measurements

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Neutrons are one of possible background candidates in rare event search experiments at an underground laboratory. It is known that they are mostly generated from (α, n) process at daughter isotopes of ^{238}U and ^{232}Th decay chains in the rock and from reactions with cosmic rays. For neutron energy spectrum, neutron rate measurement using ^{3}He-filled proportional counter with several different size of Bonner spheres and unfolding method are used.

However, there are two technical difficulties for the neutron measurement in the underground. The one is a quite low neutron rate and the other is estimation and subtraction of background rates. The one can be resolved by a long-exposure measurement [1,2]. Then, intrinsic background of ^{3}He-filled proportional counter was appeared [3], while it was not at short-exposure measurement. Those background are α-particle signals, originated from daughter isotopes of ^{238}U and ^{232}Th decay chains at mostly detector internal surface. To validate a response of α-particle in the ^{3}He-filled proportional counter, Monte Carlo simulations was performed using GEANT4 package. Detector response of α-particles from radioactive isotopes, ^{238}U, ^{232}Th, and their daughter isotopes, in the ^{3}He-filled proportional counter were simulated and compared the result with measured data. In this paper, we will discuss methods to estimate and subtract alpha backgrounds. Also, alpha activity of the detector will be estimated.

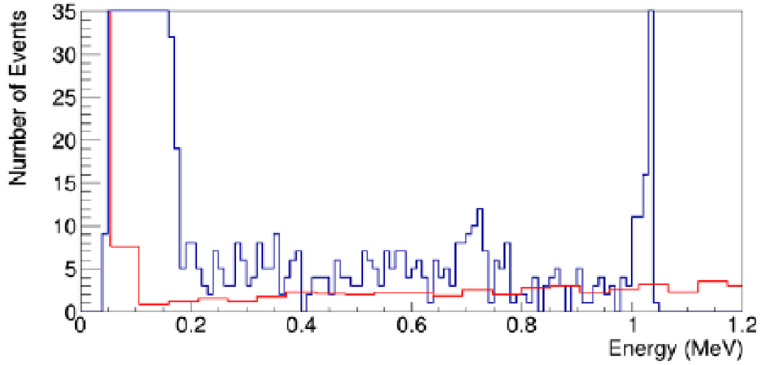

Fig. 1. Comparison of measurement results(blue) and a scaled α simulation result (red)

Keywords: Underground laboratory, Dosimetry, neutron measurement, alpha background

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PS2 (T2.3-0942)

A FPGA based coincidence module for TDCR counting system

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The triple-to-double coincidence ratio (TDCR) is one of the most useful liquid scintillation counting methods for primary activity determination of pure beta and electron capture emitters. The method requires a special counting system consists of three photomultiplier tubes (PMT) arranged in 120° geometry and complex electronics to process the signals delivered from the three PMTs simultaneously. To process all the signals simultaneously, the MAC3 coincidence module developed by the BNM-LNHB (formerly BNM-LPRI) is normally used in a primary TDCR counting system around the world. The module consists of several logic gates for processing the signals delivered by the three PMT channels as well as provide the extendable dead-time management. Instead of using the MAC3 module, we are developing the new TDCR counting system based on Field Programmable Gate Arrays (FPGAs). The FPGA will implement several logic algorithms providing the same functionalities of the MAC3, i.e., as a coincidence and dead-time unit. In addition to that, the FPGA is also programmed to provide counter and timer functions for the three channels.

The FPGA based coincidence module is developed by using a compact Reconfigurable Input/Output (cRIO) controller coupled with a configurable digital I/O interface module to receive the fast logic signals from a constant fraction discriminator (CFD). The controller is embedded with a real-time (RT) processor that guarantees all process can be completed within a defined fixed time constrain. Signals from CFD are processed on the FPGA inside the cRIO controller to give the output of single and coincidence counting rates data. The data is then transfer to the RT processor using DMA FIFO method for further data handling and analysis. All algorithm for the FPGA and RT is developed using Labview programming language with FPGA and RT modules. A computer host program was also developed for both control and monitoring the RT.

This paper described the logic algorithm applied on the FPGA, the communication and data transfer scheme between the FPGA, RT, and the computer host. The fully functioning FPGA coincidence module is then tested by replacing the MAC3 module used in the TDCR system for measuring the activity of ^{90}Sr. The result is compared with those obtained with the MAC3 module.

Keywords: *FPGA, coincidence module, tdcr*

ACKNOWLEDGMENTS
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PS2 (T2.3-0948)

Design and optimization of analog pre-filter for mixed-signal processing for portable CdZnTe gamma-ray spectrometry system

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For development of a low-cost portable CZT gamma-ray spectrometry system, analog-digital mixed-signal processing was adopted. Our analog signal processing is applied to a pre-amplifier output pulse for eliminating the aliasing effect and enhancing the energy resolution rather than digitizing the pre-amplifier output pulse directly using a low sampling rate digitizer. In this paper, the analog board for a high-resolution gamma-ray spectrometry system using low-cost and low sampling rate digitizer was designed and characterized.

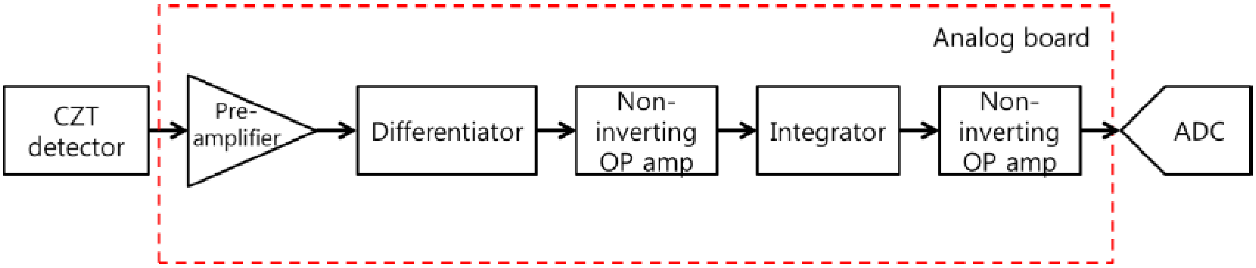

Fig. 1. The schematic diagram of the analog board

As shown in Fig. 1, our analog board consists of a pre-amplifier, a differentiator, an integrator and two non-inverting op amp circuits. Analog signal processing is composed of three steps. The first stage is for reducing pile-up by shortening the decay time of the input signal. The second stage conduct anti-aliasing at the integrator circuit. The final stage is to match the full-scale range of the digitizer using gain adjustment and offset voltage at the OP amp circuit. The rise time of the final output signal is a few microseconds so that aliasing cannot occur. For validation of our pre-filter, the energy resolution at 662 keV gamma-ray energy was measured.

PS2 (T2.3-0972)

Troubleshooting Irradiators of Gamma Service Medical Company

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The irradiators belonging to Gamma Service Medical Company have many years of exploitation and there is not any personnel properly trained in the Standart Secondary Dosimetry Laboratory (SSDL), so these facilities do not have such specialized personnel in these machines. The experiences are arranged from the perspective of Radiological Protection.

Take in to account this situation, this presentation has as a fundamental objective to provide basic acknowledge for the location of common faults to them. This documentation has its fundamental base in the practical experiences accumulated through 20 years of the author. Is important to know that the 20% of its components, generates 80% of the problems that arise (between others, the defects mainly focus on the solenoid, on LED diode lamps and on two half-power transistors of one of the control panel cards, on the proximity sensors that control the shutter's open and closed states, in the actuator and in the resistance fatigue of the metal that is in the cabinet that contains the radiation source). Although the most important deteriorations to solve is the fatigue of the metal, mainly of the arm that exerts a lever in the opening of the shutter. The experience of solenoid recovery, backed by more than six years of exploitation, is included. A solenoid costs 1900 euros.

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PS2 (T2.3-1000)

Quasi-isothermal mode operation of a graphite calorimeter under the high energy X-ray beams

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A graphite calorimeter was tried to run in the quasi-isothermal mode [1] (semi-active) under MV photon beams, which was originally developed to run in the quasi-adiabatic mode. The calorimeter was developed in 2016 [2] by the Korea Research Institute of Standards and Science (KRISS) as the primary standard of the absorbed-dose-to-water of the MV photon beams. Equivalence of the absorbed-dose-to-water standard established with the calorimeter (run in the quasi-adiabatic mode) was demonstrated by participating in comparison studies; a bilateral comparison study between the KRISS and the National Metrology Institute of Japan (NMIJ) in 2016 [3] and the key comparison study of BIPM.RI(I)-K6 run by the Bureau International des Poids et Mesures (BIPM) in 2017. Degree of equivalence of the KRISS standard is available on the BIPM key comparison database (KCDB) [4].

To run the calorimeter in the quasi-isothermal mode, the feedback circuit of the temperature control program of the calorimeter was slightly modified. The calorimeter has a core, which is surrounded with two layers of jackets. In the quasi-adiabatic mode, the temperature differences among the core and the jackets were kept constant by the feedback circuit. But in the quasi-isothermal mode, the circuit was modified so that the temperatures of each body of the core and the jackets could be kept constant. For that purpose, an electric heater (a thermistor) implanted in the core was also incorporated into the feedback circuit, so that the circuit could control the temperature of the core, too. (The electric heater used to be used for electrical power calibration of the calorimeter when it was run in the quasi-adiabatic mode.) Since the temperature control program was built on the LABVIEW platform, the modification was all made on the LABVIEW code.

When a graphite calorimeter runs in the iso-thermal mode, it is usual to extend the irradiation time to tens of minutes so that the dose rate of the beams could be determined from the moments when the core temperature is fully stabilized. This kind of approach is profitable running under the ^{60}Co gamma-rays, but not under the MV photon beams. Thus, in this study, irradiations were made shortly at 800 monitor units (MU), taking approximately two minutes, just same as when the calorimeter was run in the quasi-adiabatic mode. And an analysis model was also devised to determine the energy deposited to the core (graphite) by the ionizing radiation from the measurements.

The linear accelerator was Elekta Sysnergy® Platform and the irradiations were made at 6, 8, 10, 15, 18 MV. The energies imparted to the core by the ionizing radiation beams were determined. For the verification purpose, the calorimeter was also run in the quasi-adiabatic mode to compare the results obtained by different operation modes. Estimated standard uncertainty of the determined energy imparted to the core was approximately 0.26 %, which was the same in both operation modes. And the results from the modes agreed well within the estimated uncertainty by 0.17 %.

Keywords: graphite calorimeter, absorbed-dose-to-water, quasi-isothermal mode

ACKNOWLEDGMENTS

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PS2 (T2.3-1002)

Quality control of medical radiological equipment in Poland

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Quality control of medical radiological equipment in Poland became obligatory in 2005. Since then all radiological equipment has to fullfill minimum acceptability criteria before it can be used for diagnostic or treatment purposes to assure the best quality of offered services but also to assure proper radiological protection of patients and medical Staff. Many changes to regulation covering Quality Control and Quality Assurance of medical radiological equipment in Poland have been introduced since 2005 one of the most important being Amendment to the Regulation of the Minister of Health of February 18, 2011. on the conditions for the safe use of ionizing radiation for all types of medical exposure. The Amendment of December 2015 introduced a number of changes in the quality control of x-ray equipment, and also included new types of x-ray machines previously not specified in the Regulation, such as e.g. a dental cone beam CT scanner or digital mammograph.

The criteria for many tests have changed and some tests have been removed. New tests have also appeared. The changes included both specialist tests performed by accredited laboratories as well as basic tests performed by apparatus users.

In the amendment, the concept of digital radiology for the first time appeared as a separate discipline with its own test evaluation criteria. Quality control of auxiliary devices used in digital radiology such as medical monitors, CR disc scanners or printers has also been added. In some cases, this required the purchase of new quality control equipment.

The presentation shares the experience and problems associated with performing so called specialist and basic tests after the amendment has been introduced.

PS2 (T2.3-1010)

Establishment of Practical Minimum Detectable Activity Criteria of Tritium in Urine Sample for Nuclear Power Plant Workers in Korea

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Currently, the Korean nuclear power plants (NPPs) adopt the minimum detectable activity (MDA) criteria for tritium in urine samples as recommended by American national standards institute (ANSI) N13.30 (0.37 Bq/cc). The ANSI committee considered selecting the criteria whether or not the values presented in the published literatures, such as International Commission for Radiation Protection (ICRP) publications, were reasonable in terms of health physics and achievable for laboratory. Therefore, it is necessary to consider the same conditions to set MDA criteria for tritium at Korean NPPs what the ANSI committee did.

The probability of internal exposure due to the tritium is relatively higher in pressurized heavy water reactor (PHWR) than that in pressurized water reactor since heavy water is used as both moderator and coolant in PHWR. Thus, the tritium monitoring for workers' urine sample, using a liquid scintillator counter, mainly conducted at PHWRs in Korea. The period of the urine sample monitoring program is usually two weeks. Due to a large number of urine samples, approximately 20000 urine samples in a year, the measurement time is set as 30 seconds or 1 minute per sample. Although the measurement time is short, the annual performance test results show that all performance index is satisfied with the acceptable range. However, the MDA of tritium (0.41 - 0.88 Bq/cc) is difficult to meet the criteria of ANSI N13.30. A health physicist (HP) in Korean NPPs controls the internal exposure dose of workers following the reference level, according to the dose level, such as screening level (0.02 mSv/y), recording level (0.1 mSv/y), investigation level (1 mSv/y) and medical intervention level (16 mSv/y). If a worker's exposure dose was expected below the screening level, a HP does not perform dose assessment. If the exposure dose is expected between the screening level and the recording level, a HP records the dose. Therefore, the NPPs urine analysis system should be able to measure the concentration of tritium in urine samples that are at least corresponding to the screening level. Furthermore, the internal dose assessment method for the tritium intake recommended by ANSI N13.14 is applied to Korean NPPs. With that method, the internal exposure dose corresponds to the screening level if 1 Bq/cc of tritium was detected in the urine sample of the worker under normal monitoring conditions (14 days).

Finally, it is estimated that 1 Bq/cc, which is a reasonable amount from the perspective of health physics and achievable in a current analysis system conditions of laboratory in NPPs, can be applied as a practical MDA criteria for tritium in the urine of NPP workers.

Table 1. Performance test results during the recent three years (2017–2019)

Performance index	2017	2018	2019	Acceptable range*
Relative bias (B_r)	-0.01 to 0.15	-0.10 to 0.16	-0.02 to -0.01	$-0.25 \leqq B_r \leqq 0.50$
Relative precision (S_B)	0.18 to 0.22	0.05 to 0.11	0.03 to 0.03	$S_B \leqq 0.40$
Root mean squared error (RMSE)	0.22 to 0.24	0.11 to 0.20	0.03 to 0.04	RMSE $\leqq$ 0.25
MDA	0.62 to 0.73	0.41 to 0.72	0.57 to 0.88	MDA $\leqq$ 0.37 Bq/cc

* ANSI N13.30 (1996, 2011)

Keywords: *Internal exposure dose, Minimum detectable activity, Urine analysis*

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PS2 (T2.3-1013)

Radio-isotope Identification Using Compressed Sensing for γ-ray Spectra

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Radio-isotope identification (RIID) using γ-ray spectra has widely been required in many fields. General approaches in RIID algorithms including simple library comparisons, region of interest (ROI) and template matching showed a limitation when the spectra was obtained from multi-isotope with unstable electronics drifting. Recently, machine learning based algorithm such as artificial neural network (ANN), convolutional neural network, and support vector machine has been intensively studied for RIID to resolve the problem of previous method. In this work, a compressed sensing (CS), which is a signal processing theory for efficiently acquiring data by finding sparse solution, based classification approach was proposed as a novel algorithm for identifying multi-isotopes for a 2 inch NaI(Tl) scintillation detector. Label consistent K-SVD (LC-KSVD) algorithm was used for discriminative sparse representation [1-2]. As shown in Fig. 1, the K-SVD and the orthogonal matching pursuit (OMP) algorithm were used for dictionary learning and sparse coding. Using computed Gaussian energy broadening (GEB) coefficient from measured radio-isotopes (^{22}Na, ^{137}Cs, and ^{60}Co) spectra, MCNP6 with variable number of particle histories (NPS) was employed to produce γ-ray spectra. Training samples per each label for dictionary learning was 200. To evaluate the performance of RIID, test samples were also generated with 50 samples per each label. Considering the trade-off effect between the RIID accuracy and the computational complexity, suitable dictionary size and sparsity should be setup. The results demonstrated that a dictionary with 812 atoms showed the high accuracy (99.1%) for test samples with only 1 sparsity. It was expected that RIID with sparse representation is good for on-board applications due to its low complexity and robust to bit loss during the signal transfer.

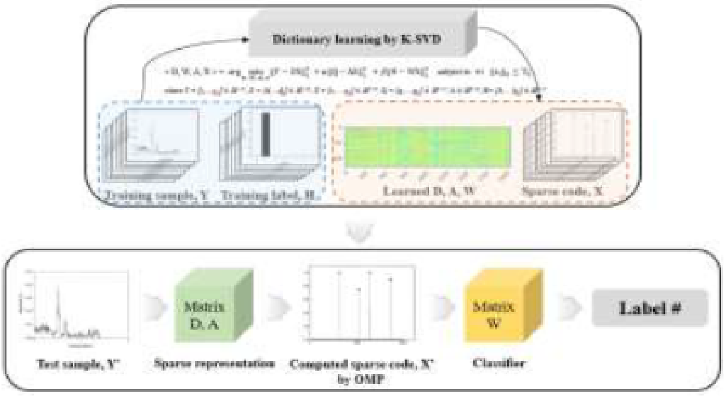

Fig. 1. Simplified diagram of the dictionary learning based RIID approach for γ-ray spectra

Keywords: Radio-isotope identification, Compressed sensing, γ-ray spectra

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PS2 (T2.3-1017)

Development of a 4π(LS)β-γ Coincidence Counting System with a movable 3-PM Beta Counter for Activity Standardization

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A new 4π(LS)β-γ coincidence counting system with a movable beta counter to vary beta efficiency has been developed at KRISS. The system incorporates three photomultiplier tubes, two NaI(Tl) detectors with 3 inch diameter located above and below the center of the three-PM plane, and an integral light-tight housing. The three PM tubes for β-counting in one plane are simultaneously movable up to 100 mm from a liquid scintillation vial, thus enable variation of β-detection efficiencies by a geometrical technique to apply the efficiency-extrapolation method. The installation of two γ-detectors allows 4π(LS)β-γ coincidence measurement to be performed in parallel for two γ-windows. The β-event is determined by counting the logical sum of three double coincidences. All the necessary electronics, i.e. analyzing AND/OR relation and logical sum, adjusting the duration of dead-time of each counting channel and coincidence resolving times, were specially designed to be fabricated in an integrated circuit for this work. Details of the detectors, the electronics and the overall movable 4π(LS)β-γ system are presented, as well as the results of investigations to assess its operating characteristics. Validation measurements have been performed with the same ^{60}Co sources as measured with the previous system [1].
The highest β-detection efficiency achieved with ^{60}Co was 95 %. The activity concentration determined with a movable 4π(LS)β-γ coincidence counter agreed with certified value within the range of uncertainties. Further results from validation measurements and the corresponding uncertainty budgets are presented.

Keywords: 4π(LS)β-γ(NaI(Tl)) coincidence counter, logical sum of three double coincidences, standardization of ^{60}Co

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This study was supported by the standards maintenance research program of KRISS and in part by the research staff exchange program of Mokwon University.

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PS2 (T2.3-1046)

Changes of KRISS primary standards by implementing ICRU report 90 recommendations

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ICRU (International Commission of Radiation Units and Measurements) has published report No. 90 (Key Data for Ionizing Radiation Dosimetry: Measurement Standards and Applications) in 2016 [1]. KRISS agreed in the 23rd biannual meeting of Consultative Committee for Ionizing Radiation (CCRI) in 2017 to implement ICRU report 90 recommendations to our primary standards. The report presents the changes of KRISS primary standards obtained by implementing recommendations given in ICRU report 90.

Related to the primary standards of KRISS (Korea Research Institute of Standards and Science), three recommendations should be considered. At first, two new correction factors (k_{ii} and k_W) are introduced for the free-air ionization chamber, which influences majorly to the standards for kV X-ray beams. New correction factors for free-air ionization chamber were calculated by simulating the initial electrons' generations using GEANT4 code [2]. The second recommendation considered is for changes of the mean excitation energies of graphite and water, and the use of the crystalline density of graphite in the stopping power evaluation. These recommendations are related with the graphite cavity chamber and graphite calorimeter, which are primary standards of KRISS for air kerma in gamma-rays and absorbed dose to water in high energy photons, respectively. The effect of the recommended mean excitation energy and adoption of the crystalline density in stopping power evaluation for graphite was studied for the KRISS graphite cavity chamber using EGSnrc code [3]. Also, the recommendations on graphite and water were adopted for the KRISS graphite calorimeter in MV photon beams to evaluate dose conversion factors from graphite to water using EGSnrc. Thirdly, considered is the recommendation for increasing the uncertainty of the mean ionization energy in air from 0.15 % to 0.35 %. Influence of the increased uncertainty in the mean ionization energy of air when combined with the existing components were discussed in detail, together with the overall changes of KRISS air kerma primary standards.

As a result, the products of the calculated k_{ii} and k_W for the free-air ionization chamber showed the maximum difference from the values given in ICRU report 90 by 0.08% for the mono-energetic photons in range of 1 keV – 300 keV. For the X-ray beams recommended by BIPM-CCRI(I), the calculated correction factors agreed with the BIPM standards [4] within 0.02 %. For the graphite cavity chamber, the changes of approximately 0.7 % – 0.8 % were observed for air kerma standards in ^{60}Co and ^{137}Cs gamma rays, which are similar to those of BIPM standards [4]. For the KRISS graphite calorimeter in high energy photon beams, the evaluated dose conversion factors based on ICRU 90 were slightly less than the evaluations with ICRU 37. However, the resulting changes are within the statistic uncertainties of MC calculations for the X-rays at 6, 10, and 18 MV.

Keywords: Dosimetry, Primary standard, ICRU 90

ACKNOWLEDGMENTS

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PS2 (T2.3-1047)

A compact and light-weight hand-held radio-isotope identification device

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We have developed our 3rd generation hand-held RIID (radio-isotope identification device) and it is the most compact and light-weight ever produced in this category worldwide.

It consists of up to a 2x2 inch organic scintillator (NaI(Tl), $CeBr_3$ or $LaBr_3$) and a 2x2 cm^2 solid-state sensor for Gamma and Neutron detection, respectively. The key feature of the device includes sourceless auto-stabilization using natural background, real-time reach-back service, a high resolution camera for appending images or video, multi-channel GPS for accurate location information, replaceable battery pack for up to 14 hours of operation, a gyro-sensor for isotope location finder, and more.

It is designed to address a growing demand for fast, accurate isotope identification. The device reduces the burden on the operator by leveraging state of the art communication protocols to quickly relay information to a management team. The light weight design also limits fatigue in long operational scenarios or enables easy transport of your identifier in a belt holster or lanyard. In addition to ANSI compliant spectroscopic results and alarming, the user has quick access to the onboard camera for appending video or images to the spectroscopic reports.

One of the most advanced feature of the unit is its connectability to external devices. The device allows to connect to external prove, SIM card for 4G/LTE network or GPS device easily with the micro USB port.

Its energy resolution is less than 7% and 4% at 662 keV for NaI(Tl) and $LaBr_3$, respectively and linearity of energy is within 1% by real-time linearization by firmware.

***Keywords**: RIID, Potable device, Radiation direction*

PS2 (T2.3-1067)

An Effective Sealing Method to Prevent Radon Leakage for Measuring ^{226}Ra Using Gamma-ray Spectrometry

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The determination of ^{226}Ra in environmental samples using gamma-ray spectrometry has been based on the detection of gamma-rays of the radon progeny ^{214}Pb and ^{214}Bi. This method requires an ingrowth period of minimum 20 days, during which the sample has been tightly sealed to ensure secular equilibrium between ^{226}Ra and its progeny. Various measurement containers such as aluminum cans, plastic bottles, glass bottles and Teflon vessels have been used to prevent radon leakage.[1, 2] Several studies suggested the different sealing methods for radon tightness. Vacuum packaging of the measurement container in a sealed aluminum bag is one of the effective method.[3] In addition, it may be a solution to seal the lid of the measurement container with a sealant such as plastic foil, paraffin, epoxy adhesive, aluminum tape and foil.[4] When using these methods, the measurement container should to be filled with the samples completely in order to avoid effective efficiency change due to radon accumulating in the void volume. Therefore, it may be difficult to use these methods when the sample amount is insufficient.

In this study, we presented an effective sealing method to eliminate the empty space on top of the sample regardless of the sample amount. The sealing method is as follows. The sample is filled into a cylindrical polystyrene container having a diameter of 60 mm, a height of 54 mm, a wall thickness of 1.5 mm. The top of the sample is covered with an acrylic of 60 mm in diameter and 10 mm in thickness. Then seal the gap between the measurement container and the acrylic with a sealant. To investigate the time of build-up of the progenies of ^{222}Rn and the tightness of the container IAEA reference materials (IAEA 434 and 448) were used. Five samples were sealed with silicon sealant and three samples sealed with butyl rubber sealant. Sealed samples were measured periodically using a high purity germanium detector. After sufficient build-up time, the measured ^{226}Ra activity concentration was compared to the reference value using zeta score test. As a result, results for all of the prepared samples are in statistical agreement with the reference value. This sealing method does not requires separate efficiency calibration and can be used regardless of the sample amount.

***Keywords**: Gamma-ray spectrometry, Sample preparation, Radium-226.*

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PS2 (T2.3-1069)

Development and verification of the dual-OSL chip reader and algorithm

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For several radiation practices, such as radiation workers operating radioisotopes in nuclear medicine department, hand exposure is inevitable and is usually obtained higher radiation doses. Therefore, wearing the ring dosimeter on the finger to monitor the hand (extremity) dose is frequently used. In the previous study, the OSL-ring dosimeter was designed and manufactured which dosimeter is composed of two OSL chips. In order to avoid fading during reading the OSL-ring dosimeter and to consider reading automation; this study developed a new type of reading instrument which could be used to read the OSL-ring dosimeter in a single process. Considering the error due to the realistic radiation practice, the angular dependence of the OSL-ring dosimeter was tested and the algorithm was designed in three different modes to minimum the error. In this study, the OSL-ring dosimeter system including the reader and algorithm was verified by means of the irradiated test using standard X ray, gamma ray beams in the National Radiation Standard Laboratory of Institute of Nuclear Energy Research in Taiwan. The tested results of the new system are presented in this study. In conclusion, reading automation and accuracy of ring dosimeters could be improved in this study.

Keywords: Ring dosimeter, extremity dosimeter, OSLD

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PS2 (T2.3-1073)

Establishment of Radiation Quality of Narrow Spectrum Beams Used in Secondary Standard Dosimetry SSDL According to ISO 4037-1:1996

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This study aimed to established the radiation quality of narrow-spectrum beam NS of X-ray Irradiator in the National Secondary Standards Laboratory SSDL- CNRP- Ministry of Health- Morocco according to ISO 4037-1:1996. The ISO standard method ISM is describe the requirements and the procedures to determination the inherent filtration for X radiation qualities, first and second half value layer (1^{st} HVL, 2^{nd} HVL) and homogeneity coefficients (h) by HVL measurement method. Many parameters have to adjust in order to study the X-ray beam quality, High-voltage a cross the X-ray tube, current intensity, thickness and total filtration, it is also depend in the properties of the target within tube and other. This characteristics of the X-ray beams are determined for the narrow spectrum series NS40 to NS300 . The results show good agreement between the ISO and the experimental results for all energy spectrum series of the x-ray tube of the narrow- spectrum series. Also, the homogeneity coefficients h was between 0.75 to 1,00 according to the ISO standard. The minimum values for 1^{st} HVL and 2^{nd} HVL were 0.47% and 0.12% respectively and the maximum were 4.80% and 4.86% respectively while h was between 0.91 to 1.00. The radiation quality of narrow-spectrum was established and results was in the range according to ISO less than 5% .

Keywords*: SSDL, X-Ray irradiator, ISO 4037-1, Narrow-spectrum NS, radiation quality, HVL, homogeneity coefficient h.*

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PS2 (T2.3-1074)

Comparison and Optimization of the Analytical Method for ^{210}Po Determination in Environmental Samples Using Alpha-spectrometry

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Polonium-210 is an alpha emitter (E_α; 5.304 MeV, half-life; 138.4 days) in the natural ^{238}U decay series nuclide and is produced from the decay of ^{222}Rn gas. ^{210}Po is highly toxic and known as a major radionuclide that has an effect on internal exposure.[1] ^{210}Po is widely distributed in all environments, so enters the human body through ingestion, inhalation or absorption. The alpha-spectrometer have been mainly used for ^{210}Po analysis, which involves various pretreatment processes such as decomposition, pre-concentration, chemical separation, and source preparation. [2, 3] In particular, soil and biological samples are decomposed with various acids such as HF, H_2O_2, HNO_3, HCl to extract the nuclides inside the sample matrix. For pre-concentration of target nuclide, co-precipitation with ferric hydroxide or manganese are mainly used. At this time, various interfering nuclides are co-pricipitated with target nuclides and their presence reduces the recovery rate during auto-deposition of polonium on the silver disk. Thus chemical separation such as solvent extraction, ion exchange and extraction chromatography are required before α-measurement. Although several studies have been conducted for ^{210}Po analysis, further comparison and optimization studies for each pretreatment process are still required in various environmental samples.

In this study, various pretreatment processes for analysis of ^{210}Po in environmental samples such as soil and biological samples were validated and optimized. The co-precipitation methods with ferric hydroxide and manganese were compared and extraction chromatography using Sr-resin was used to observe the behavior of Polonium to obtain the optimum conditions. Then, using an auto-deposition kit manufactured in our laboratory, experiments were carried out under different conditions such as time, temperature, pH, and sample volume to increase the recovery rate. Prior to validation with CRM(Certificated Reference Materials), water samples containing ^{208}Po and ^{209}Po tracers were used, and each tracer was spiked before co-precipitation and auto-deposition to confirm the recovery rate of each analysis process. Using the same method, the CRMs were analyzed and the activity concentration in the CRM sample was calculated based on the recovery rate of tracer, and the result is compared with the certificate or reference value.

***Keywords**: Polonium-210, alpha-spectrometry, Sr-resin*

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PS2 (T2.3-1077)

Development of irradiation phantom for biological research and dosimetry

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In biological research, irradiation has been used on cells, blood, mice, and food for anticancer therapy research, the latest cancer research, immune response induction, tumor bed effect, and dose evaluation. However, no dose evaluation has been made for tubes or flasks of different sizes depending on the irradiation sample, and accurate dose evaluation is required to obtain reliable irradiation dose measurements.

In this study, we fabricated an irradiation phantom for use in biological research for the irradiation of biospecimen (cells, blood, mice) [1]. To evaluate the dosimetric characteristics of the phantom, irradiation was performed using a cobalt irradiator (GBX200, Best Theratronics, Canada) containing ^{60}Co. The dose rate was measured using an ionization chamber. To evaluate the dosimetric characteristics of the phantom, percentage depth dose (PDD), output factor, and beam profile were measured [2]. In addition, for dose evaluation of the rotation mode during the irradiation of a circular tube, an alanine dosimeter was irradiated, and the dose was measured using an electron paramagnetic resonance (ERR) spectrometer.

The results of the percentage depth dose measurements obtained using the ionization chamber and the alanine dosimeter showed a 0.7% difference. Furthermore, when the output factor and beam profiles were considered and the biospecimen were irradiated using an irradiation plane of 30×25 cm^2, the dose distribution was uniform at all five measuring points. In addition, the dose comparison for the rotation mode using the alanine dosimeter showed a maximum difference of 6.8%.

Thus, this study analyzed the dosimetric characteristics of biospecimen irradiation and evaluated its usefulness through the development of an irradiation phantom for biological research and dose measurement. It will also be able to provide reliable information regarding irradiation for researchers conducting biological irradiation.

Keywords: *Irradiation phantom, Dosimetry, Alanine/EPR*

ACKNOWLEDGMENTS

This work was supported by the Dongnam Institute of Radiological & Medical Sciences (DIRAMS) grant funded by the Korea government (MSIT) (No. 50496-2020)

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PS2 (T2.3-1087)

The Simple Discrimination Methods for Tritium and Carbon-14 in Aqueous Samples using a Liquid Scintillation Counter (LSC)

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In Pressurized Heavy Water Reactors (PHWRs), tritium and carbon-14 in urine samples are analyzed using a Liquid Scintillation Counter (LSC) to evaluate the internal exposure dose. In Korea tritium is analyzed periodically, and carbon-14 is further analyzed when ingestion is suspected [1]. Unlike environmental samples, these urine samples are directly analyzed without any pretreatments. Therefore, if tritium and carbon-14 are simultaneously present in the same sample, this direct counting method can lead to false measurement results because the energy ranges of these two nuclides overlap each other. In this study, we developed the discrimination methods for the tritium and carbon-14 that can be applied to any analysis conditions for LSC.
In this study, we developed the simple verification methods for tritium and carbon-14 measurement results that can be used in any analysis conditions for LSCs by comparing count-rate or activity in their each counting region. As the methods to verify the reliability of the measurement results, the ratio of the counting ratios of the each counting regions (A, B and C) and the radioactivity of each region were compared. As a result, the CPMA / CPMC ratio was more than 95%, and the CPMB / CPMC ratio was less than 2% in the condition 1 when only tritium was present in the sample. In only the presence of carbon-14 in the sample, the CPMA / CPMC ratio was more than 70%, and the CPMA / CPMC ratio was less than 4% in the condition 2. Moreover, when only a single nuclide was present in the sample, the radioactivity of each counting region A, B, and C had the same result within 2%.

Keywords: *Liquid scintillation counter, Single Label DPM, tritium, carbon-14, QIP, Counting Region*

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PS2 (T2.3-1092)

Development of a Gamma Detector with SrI_2 Scintillator and SiPM Sensor for Tomographic Imaging with High Resolved Energy Discrimination

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Gamma ray detection with high energy resolution is important for the appropriate monitoring and quantitation of medical radiation and radioactive waste. Among the many kinds of radiation monitoring devices, detectors using SrI_2 scintillator and traditional photomultiplier tube have been used in the monitoring of environmental or radioactive waste radiation, because of the high energy resolution, but there has been no study of tomographic imaging using SrI_2 although tomographic imaging can provide more information of quantitative radiation distribution as well as photon energy. We developed a prototype imaging camera using SrI_2 and silicon photomultiplier (SiPM) for gamma imaging. SrI_2 crystal (C&A Co., Japan) with pixel array (8 by 8, 3 × 3 × 10 mm3/pixel) was used for 1:1 matching to multi-pixel photon counter (Hamamatsu Photonics, Japan). A parallel hole typed collimator for high energy of 511 keV was designed and manufactured by 3-D printing. A set of commercial devices (AiT Instruments, VA) was used for readout and data processing of acquired signals. Point and line sources of F-18 were used for detector calibration and imaging. An in-house device for rotating source was used for tomographic acquisition of two-line sources. The pixel-by-pixel sensitivity map of camera detector could be obtained by segmentation process. The energy spectrum of each crystal pixel was obtained. After the calibration of peak position of energy spectra, the mean value of energy resolution was improved to 7.3%. In the acquired projection data, the both lines of parallel line sources were shown well. The projection data of 60 views over 360 degrees was reconstructed to tomographic image using MLEM algorithm (20 iterations). In the reconstructed image, the shape and position of the line sources were observed clearly. The value of spatial resolution was 3.3 mm. The preliminary results show the SrI_2 can be applied to image detector and the gamma camera using SrI_2 and SiPM have a potential of radiation imaging of high energy resolution that would be useful in the monitoring radiation exposure.

Keywords: SrI_2 scintillator, SiPM sensor, gamma camera

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PS2 (T2.3-1098)

Fully Automated Evaluation System for Medical Gamma Probes Using LabVIEW

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Gamma probes have been widely used in surgery oriented to locate the sentinel lymph nodes in breast cancer and malignant melanoma during the surgery. In general, Full width half maximum (FWHM) radial sensitivity distribution at 30cm distance, FWHM spatial resolution at 1cm distance, Max. sensitivity of a point source, Max. sensitivity outside measurement-field(leak sensitivity), ratio of leak sensitivity to max. sensitivity should be quantitatively measured, evaluated and analyzed as key indicators to verify the probe's performance and features. Various setups of these experimental procedure are required a lot of time, so that the experimenter risks a lot of exposure.

In this paper, the fully automated system based on LabVIEW was implemented to minimize the mentioned problem. The system can fully measure the five key performance indicators mentioned earlier and display the results in real time. Fig. 1 shows a program capture of the apparatus. It has a real-time graph display of the measured results, the monitoring of signal fluctuation of the acquired data, data storage, and the table display of results according to the distance and angle between the radiation source and the detector. In addition, an intuitive indicator for monitoring the state of the hardware was implemented separately.

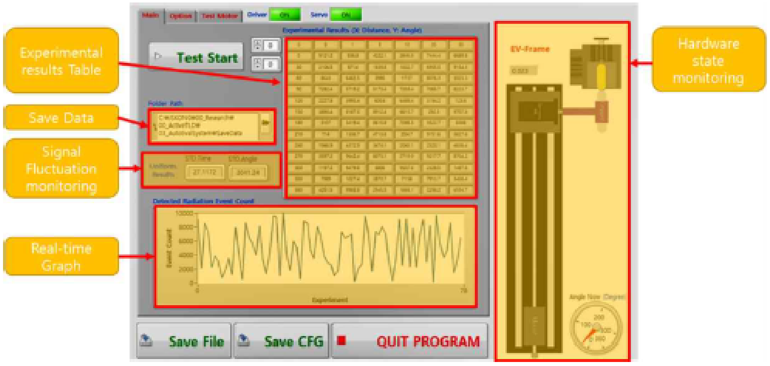

Fig. 1. GUI program for medical gamma probes

***Keywords**: Gamma probe, evaluation system, LabVIEW, Automation*

ACKNOWLEDGMENTS

Acknowledgments can be placed here if needed. (left alignment)

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PS2 (T2.3-1121)

Comparison exercises as a tool for harmonization of calibration methods of the surface contamination monitors in Latin American region

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Radioactive contamination can arise from several activities, to evaluate these contaminations it may be necessary to use monitors. Surface contamination is evaluated in terms of activity per unit area and the limits are based on the international recommendations, but the reading of the monitors are mainly done in counts rates. The calibration of the monitors is a crucial step on evaluation, nevertheless it is impracticable to perform it for every radionuclide. As an alternative it is possible to test the energy response characteristics of the monitors for alpha-, beta- and photon emissions using reference sources that do not necessarly represent the surface to be monitored. The radionuclide-specific factors of the monitor can be derived from the interpolation of all test results based on the known decay data. This measurement method is described in the ISO 7503 standards recently updated.

From 2013 to 2019 two regional comparison exercises on calibration methods of contamination monitors were organized in the Secondary Standard Dosimetry Laboratory from the *Centro de Protección e Higiene de las Radiaciones* of Cuba. The comparisons were sponsered by International Atomic Energy Agency as a part of the activities of the projects RLA 9066, RLA9075 and RLA9085. The objectives of the comparisons were to assess the implementation level of the calibration methods in the Latin American region in order to propose the way for its harmonization. Nine laboratories from Argentina, Brasil Chile, Costa Rica, Cuba, Ecuador, Mexico, Nicaragua and Peru participated in the comparisons. The Cuban laboratory participated as the pilot in both comparison exercises.

The number En from the ISO-IEC 17043 was selected to evaluate the performance of the laboratories. In total 20 of 40 individual results were evaluated as satisfactory. In most case, the laboratories use alpha and beta sources for calibration. Only four measurements were done with photon sources and three resulted unsatisfactory. It was concluded that the calibration methods of the contamination monitors and uncertainty estimations still need to be improved within the region.

Keywords: calibration, surface contamination, monitor

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The authors gratefully acknowledge the important participation of the colleagues from the nine laboratories of the region and especially acknowledge the contribution of Ms. Gladys M. López-Bejerano, Mr. Manuel López-Rodríguez, Mr. Leonides Bravo Leyva and Mr. Andy L. Romero Acosta during organization and measurement process in the *Centro de Protección e Higiene de las Radiaciones*.

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PS2 (T2.3-1197)

3-D mapping of air dose rate by combining freely moving survey meter and Structure from Motion

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The Fukushima Daiichi Nuclear Power Station (FDNPS), operated by Tokyo Electric Power Company Holdings, Inc., suffered a meltdown accident due to the Great East Japan Earthquake on March 11, 2011. In order to carry out the decommissioning work of this facility smoothly and efficiently, it is extremely important to understand the distribution of air dose rates and radioactive substances in the radioactive work environment. In this report, the author presents a technique for 3-D mapping of the dose rate in the work environment.

Figure 1(a) shows the measuring device used for the demonstration experiment, which is a simple setup that combines a video camera (SONY, FDR-AX100) and a survey meter (Chiyoda technol corp., RADEYE PRD-ERJ). In the experiment, the operator carried these devices while walking around the work environment. Still images were extracted from the captured video and processed using via Structure from Motion (SfM) software (Agisoft LLC., MetaShape Professional Version 1.5.5 build 9097 (64 bit)). SfM builds a 3-D model of the work environment from multiple photographs and estimates the camera's self-position when each photograph was taken. Here, the dose rate at each camera position can be mapped by synchronizing the photographing time of each photograph and the time-stamp of the dose-rate data acquired by the survey meter. Figure 1(b) shows the 3-D model of the experiment environment reconstructed by SfM. Here, the continuous points in the figure are the movement trajectory of the devices, and the color of each point shows the value of the dose rate acquired by the survey meter. The dose rate at the point near the installation location of the ^{137}Cs-radiation source is higher than that at other points. By displaying the dose-rate data in the 3-D model in this way, the worker can easily visually recognize the dangerous place, and it will lead to the reduction of the exposure dose. In the presentation, the relationship between the dose rate and the distance toward the radiation source from each point on the movement trajectory, and the method of estimating the source position will be discussed in detail.

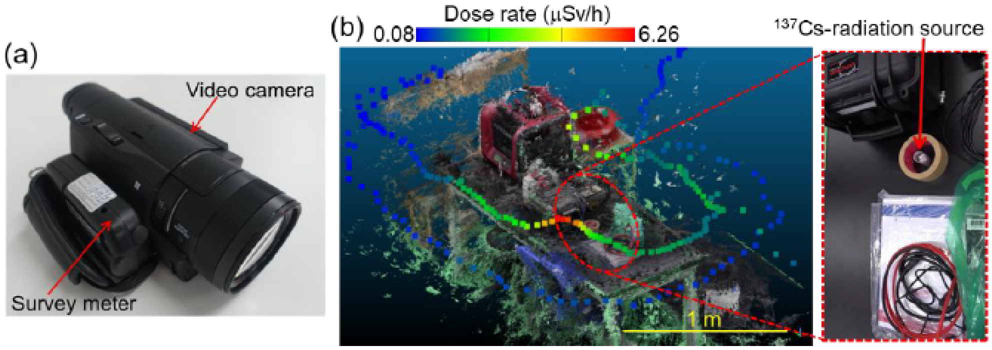

Fig. 1. A photograph of the measuring device (a) and the 3-D model of the experiment environment constructed via SfM (b). The values of the dose rate measured with the moving-survey meter are mapped on the 3-D model. A 9.7 MBq-^{137}Cs source is placed on the laboratory desk, as shown in right photograph.

***Keywords**: dose-rate mapping, 3-D mapping, Structure from Motion*

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PS2 (T2.3-1200)

Pulse Shape Discrimination of Neutrons from Gamma-ray Using 3D-Printed Plastic Scintillator with ^{6}LiF

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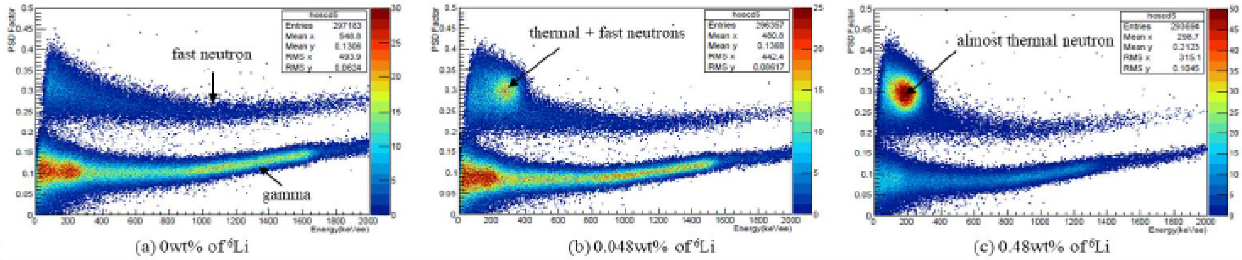
Separation technique such as pulse-shape discrimination (PSD) for neutron detection is necessary, because neutron fields are typically present with gamma-ray. Organic scintillators, such as plastic scintillators, are capable PSD using different decay times depending on particles. Plastic scintillators can be manufactured by 3D-printing technique based on digital lighting processing (DLP). The advantages of the technology are fast and simple production, low cost, and customizing easily. In previous study, it is confirmed that 3D printed scintillators with high-concentration 2,5-diphenyloxazole (PPO) was capable of fast neutron/gamma PSD [1].

In this study, 3D printed scintillators, loaded with ^{6}LiF, were produced to simultaneously measure thermal and fast neutrons using hybrid PSD. ^{6}Li has advantages such as reasonable neutron capture cross-section, relatively high energy released in the capture reaction, and the absence of gamma-ray in the final products containing charged particles [2]. The scintillators, which have diameter of 2.54 cm and thickness of 1.27 cm, were connected to PMT (Hamamatsu-H6410) and FADC (Notice-NGT400). Data acquisition (DAQ) system based on ROOT framework was operated in PC through Ethernet. For neutron irradiations of this work, ^{252}Cf source (88.3 μCi), which shielded with 5 cm of lead and moderated with high density polyethylene of 11 cm, was used. The scintillators were evaluated using figure of merit (FOM) for the capability of PSD in energy region from 0 to 1,000 keV_{ee} (keV electron equivalent). The reasonable value of FOM for well separated Gaussian distributions is more than 1.27. The results of PSD respectively are shown Fig. 1 below. PSD factor means ratio of body and tail in spectrums. FOM of scintillator (0wt% of ^{6}Li) was 1.39 ± 0.015 for fast neutron/gamma PSD. FOMs of scintillator (0.048wt% of ^{6}Li) were 1.75 ± 0.035 and 0.49 ± 0.031 for fast neutron/gamma and thermal/fast neutrons hybrid PSD respectively. FOM of scintillator (0.48wt% of ^{6}Li) was 1.9 ± 0.029 for thermal neutron/gamma PSD. It is expected that the 3D-printed scintillators which are comparable to the existing commercial detectors, can be developed through continuous research and optimization.



(a) 0wt% of ^{6}Li (b) 0.048wt% of ^{6}Li (c) 0.48wt% of ^{6}Li

Fig. 1. PSD capacity change with ^{6}LiF concentration

Keywords: Pulse-shape discrimination (PSD), Plastic scintillator, 3D-Printing

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PS2 (T2.3-1206)

Development of MAGAT gel phantom for 3D star shot analysis of MR-Cobalt machine

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In this study, MAGAT gel phantom was developed and evaluated for 3D star shot analysis of MR-Cobalt machine, and minimum tangent circle radius (MTR) was compared with those of EBT3 films. The gel phantom consist of 5% weight per weight (w/w) gelatin, 6% w/w methacrylic acid, 1 mM tetrakis hydroxymethyl phosphonium chloride, and 89% w/w water. The gel phantom was irradiated with 1.05×27.3 cm^2 fields at angles of head1 at 25°, 115°, 335°, head2 at 160°, 240°, and head3 at 70°, 210° and 300°. The MR image of the gel phantom was acquired by 0.35 T MRIdian. A true fast imaging was used with a steady state precession sequence, yielding a T2/T1-weighted contrast. The resolution of image was 1.5×1.5 mm^2, with an imaging time of 128 seconds [1]. The films were irradiated with 1.05 × 27.3 cm^2 fields using four plans; plan1 at 0°, 72°, 144°, 216°, 288°, plan2 at 18°, 90°, 162°, 234°, 306°, plan3 at 36°, 108°, 180°, 252°, 324°, and plan4 at 54°, 126°, 198°, 270°, 342°. The films were scanned using Epson 10000XL flatbed scanner with 0.08×0.08 mm^2 resolution. We analyzed the star shots acquired by MAGAT and film by using MATLAB. The mean MTRs were 2.33 ± 0.21 mm for 11 slices of MAGAT and 1.00 ± 0.10 mm for four EBT films. In conclusion, the gel phantom can be used usefully by QA tool of MR-cobalt machine because it can be analyzed in 3D star shot without changing the set up compared to the film. However, MRTs of MAGAT were larger than tolerance of TG-142. It might due to relatively large spatial resolution of TRUFI image. Therefore, the superior image spatial resolution is required. The gel will be evaluated using another MR sequence with 1.0×1.0 mm^2 resolution.

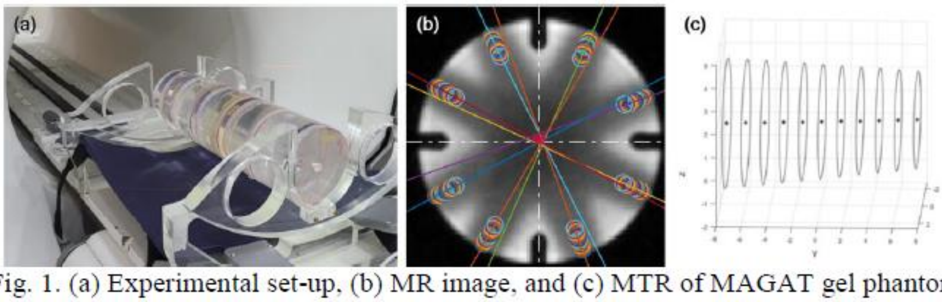

Fig. 1. (a) Experimental set-up, (b) MR image, and (c) MTR of MAGAT gel phantom

Keywords: MR-cobalt machine, Polymer gel, Star shot

ACKNOWLEDGMENTS

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PS2 (T2.3-1226)

Radiation Hardness Test of a Silicon Detector for In-Containment Coolant leakage Detection System

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A radiation hardness test of a silicon detector, which will be used to detect a coolant leakage, and be installed in a containment building of nuclear power plant, was conducted and the result was discussed. A coolant leakage detection system consists of a humidity detector, a gamma-ray detector, and a beta particle detector, etc. The detector that was tested is a beta particle detector, composed of a silicon sensor and preamplifier, which will be shielded by a 5 cm lead cylinder and be installed in the annulus zone of a nuclear power plant's primary system. A silicon sensor and a preamplifier were irradiated about 1 kGy and 1.1 kGy, respectively by ^{60}Co gamma radiation source equipped in a facility of Korea Atomic Energy Agency Advanced Radiation Technology Institute (KAERI ARTI) of the Republic of Korea. The dose had been calculated based accumulative absorbed dose on preamplifier for 60 years in an annulus zone of a nuclear power plant using Monte Carlo N-Particle (MCNP) simulation. Data of background radiation had referred Final Safety Analysis Report (FSAR) of a nuclear power plant in the Republic of Korea [1].

It is known that leakage current increases linearly with fluence in irradiated silicon sensors, and random fluctuations that inevitably occur in the leakage current will tend to obscure the small signal current [2,3]. In the case of a preamplifier, high dose radiation can make noise higher as well. To evaluate an influence on silicon sensor and preamplifier by gamma exposure, spectra by irradiated silicon sensor and preamplifier were analyzed, respectively, using an un-irradiated preamplifier and silicon sensor, which are identical with irradiated ones. The most remarkable influence on each component by gamma irradiation was that noise was increased. Table 1 shows counts rate of noise before and after gamma irradiation to each component, silicon sensor and preamplifier. A silicon sensor has more contribution to noise increase than a preamplifier in the environment of gamma irradiation, showing the noise increase of about 773 %, and 18.1 % for irradiated silicon sensor and preamplifier, respectively. Based on the result, a valid pulse height level for background discrimination was suggested.

Table 1. Counts rate before and after gamma irradiation

Components	Counts rate (before irradiation)	Counts rate (after irradiation)	Absorbed dose
Si sensor	3338.1 cps	29151.5 cps	1 kGy
Error	Reference	773 %	-
Preamp	2758.3 cps	3258.5 cps	1.1 kGy
Error	Reference	18.1 %	-

Keywords: Irradiation test, Silicon detector, Leakage current

ACKNOWLEDGMENTS

This work was supported by Korea Institute of Energy Technology Evaluation and Planning(KETEP) grant funded by the Korea government (MOTIE)

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PS2 (T2.3-1227)

Considerations of Calibration Interval of Radiation Survey Meters for Keeping the Accurate Monitoring Ability

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Many types of survey meters produced by different companies have been used at radiation-related research area and industrial facilities. It is very important procedure to calibrate survey meters regularly for keeping those accurate monitoring ability. In Korea, general ‘6 month rule’ had been requested by the authority for a long time. Recently the periods which are warranted or assured by manufacturers are used as another decision standard.

We have performed the calibration process of lab-own survey meters by officially-qualified organization last a few ten years. 21 X-ray/Gamma ion chambers, 6 neutron rem counters, 3 beta counters, 6 proportional counters and others have been calibrated every six or 12 months for last 10 years. The variation of calibration factor of all survey meters was investigated. The optimum calibration interval was studied. The variation levels are in 10 %, which is also in the acceptable range of qualified calibration. Based on the statistical estimation, the variation of calibration factors shows longer period than values suggested by each manufacturer. In this estimation, the statistical uncertainty of calibration is also considered. The records of more than 36 survey meters are introduced and the optimal access to determine the calibration interval is suggested.

***Keywords**: Calibration Interval, Survey Meter, Calibration Factor*

PS2 (T2.4-0107)

Local Diagnostic Reference Levels for Digital Mammography: Two Hospitals Study in Northwest, Nigeria

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Optimisation technique remains a key factor in reducing radiation dose to women during mammography procedure. Thus, in this study, local mammography DRL is established for the purpose of achieving patient dose reduction and acceptable image quality based on the ALARA principle. A total of 140 patient data for cranio-caudal and medio-lateral oblique projections were collected and analyzed. Demographic, exposure and dose information was extracted from the digital readouts of the DICOM header stored in the mammography workstation. Exposure parameters and dose data recorded were as follows: compressed breast thickness, kVp, mAs, exposure mode, and MGD values. The result was analyzed using SPSS v.16.0, and DRL was established at the level of 3rd quartile value based on the ICRP guidelines. The established DRL for CC and MLO was found to be 2.31mGy, which compared well with most values reported in the literature, and also, within the European recommended dose levels. Furthermore, the CBT and age were found to correlate positively with the MGD. However, the MGD for manual exposure mode is significantly higher compared to that of the automatic optimisation parameter mode. In conclusion, the locally established DRL align with the recommended European DRLs which is an indication of good local practice. However, there is a need for continuous dose monitoring and image quality assessment in accordance with the ALARA principle to consistently maintain lower patient exposure.

Keywords: Mammography, DRL, MGD

PS2 (T2.4-0731)

Taurus: A Successor to IMBA for Occupational Internal Dosimetry

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Taurus is a new software package that performs intake, bioassay and dose calculations for workers with potential internal radionuclide exposures, using the methodology recommended by the International Commission on Radiological Protection (ICRP) in the Occupational Intakes of Radionuclides (OIR) series of reports (Parts 1-4)[1,2,3,4]. It has been developed by the Internal Dosimetry Group at Public Health England's Centre for Radiation, Chemicals and Environmental Hazards (PHE-CRCE), and is the successor to the widely used Integrated Modules for Bioassay Analysis (IMBA) software[5] previously developed by the group. Unlike IMBA, Taurus is based on PHE-CRCE's internal dosimetry computer code Pleiades[6] (Fell T.P. et al, 2007) which has been used for the calculation of reference dose coefficients and bioassay quantities published in the OIR series of reports. While having an almost entirely new computational basis, Taurus also reflects user feedback received on IMBA over the years, so it has a graphical user interface (GUI) that is similar but much simpler (having just one main GUI window instead of IMBA's three) with the aim of making it quicker and easier to use.

Taurus enables the user to:

- predict bioassay quantities (e.g. urinary excretion or *in vivo* measurements of radionuclide activity), at pre-defined or user-specified time-points, and/or the effective dose and equivalent organ doses, which would arise from unit, or user-defined, intake(s), of any of the radionuclides contained in ICRP OIR Parts 1-4, following, acute and/or chronic, inhalation, ingestion or injection exposure(s);
- estimate the intake of activity, from single or multiple exposures, and resulting doses (for the same range of radionuclides and exposure types as above) from measurements of activity in the body and/or excreta using maximum likelihood fitting;
- generate publication quality plots of bioassay predictions, or measurements and the maximum likelihood fit to them, using the DPlot graph software[7].

Keywords: Dosimetry, software, bioassay

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* Note: ICRP OIR Part 4 is currently in press and once it is published a complete reference will be given

PS2 (T2.4-0848)

Emergency OSL dosimetry – search for new materials

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Along with the civilization changes there is a growing awareness of risks due to the effects of improper use of radioactive sources. Accidental dosimetry is the developing area due to the need of radiological safety assurance of society and estimation of the potential doses in the case of radiation incident. While the progress has been made in the application of thermoluminescence (TL) and optically stimulated luminescence (OSL) techniques to components of mobile phones and other electronic devices, their suitability in real emergency situations was put into question, e.g. due to reluctance of persons to part with their phones, as well as time consuming sample preparation. For that reason other, more suitable materials are still sought to be personal dosimeter during a radiological accident.

The aim of this work was to investigate the OSL properties of materials likely to be found in personal carried on bags, like commercial pharmaceuticals or popular chewing gums (placed in bags and protected from a sunlight exposure). The measurements of 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. For these materials fading correction should be applied.

The investigated materials are supposed to be very promising alternative for dose estimation during radiological accident.

Keywords: Accidental Dosimetry, Optically Stimulated Luminescence

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PS2 (T2.4-0879)

Protocol of Fingernail-EPR Dosimetry for Retrospective Assessment of Localized Exposure

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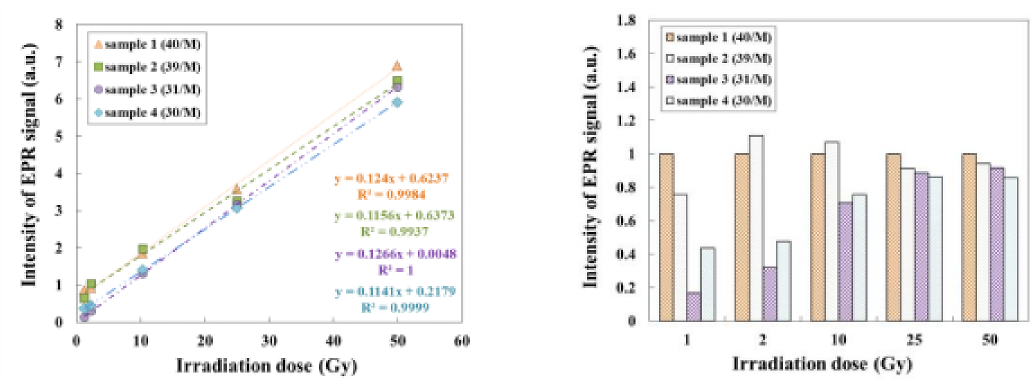
In a radiation accident situation, human tissues such as tooth enamel, hair and nail are special interests for retrospective dosimetry. For localized exposure to hands, in case of source handling accident, it is difficult to evaluate the dose distribution by calculations[1)]. However, the free radical generated in fingernail can be detected by electron paramagnetic resonance (EPR) spectrometer[2)]. Fingernail has advantage of easy collection of samples and sensitivity against radiation. Human fingernails exhibit three kinds of EPR signal: a radiation induced signal (RIS), a mechanically induced signal (MIS), and a background signal (BKS). The each of EPR spectra of fingernail are overlapped with identical and similar g-factor. Therefore, the characteristic of fingernails should be considered for analyzing the exposed dose in fingernail. The relative standard deviation (RSD) of BKS was calculated by 20 samples collected from 20 donors. MIS fading was evaluated according to time after cutting. In addition, RIS characteristic of individual dose response, fading, minimum detectable dose (MDD), and minimum mass for EPR measurement were evaluated in this study.
Figure 1 shows the RIS characteristic of individual dose response according to irradiation doses. Figure 1(a) shows the linearity between irradiation dose and intensity of EPR signal. These results show that fingernails have a good linear relationship for irradiation doses ranging from 1 to 50 Gy. Regression curves were obtained using linear fitting, and the coefficients of determination (R^2) were above 0.99 for each set of fingernail samples. Figure 1(b) shows the individual response according to the irradiation doses. The results, normalized by EPR signal of sample 1, show that the standard deviation of the intensities decreased as the irradiation dose increased. It was found that the RSDs from 1 to 50 Gy were 61.66, 53.26, 20.29, 6.57, and 6.40% respectively. These results of characteristic of fingernail will be applied in protocol of fingernail-EPR dosimetry.



(a) (b)

Fig. 1. The individual dose response of fingernail according to irradiation doses: (a) the linear relationship between the irradiation doses and the intensity of the EPR signal, and (b) the normalized intensity of the EPR signal according to the irradiation doses.

***Keywords**: Radiation accident, Retrospective dosimetry, EPR dosimetry, fingernail*

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PS2 (T2.4-0890)

Development of a dose conversion program for emergency dosimetry using luminescence materials

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In a radiation emergency scenario, a fast triage for exposed person is one of priorities. Recently, thermoluminescence (TL) / optically stimulated luminescence (OSL) technique on luminescence materials in a mobile phone are widely investigated due to their fast dose estimation [1]. Moreover, several studies calculating dose conversion coefficient (DCC) were carried out to provide a human body dose directly from a mobile phone dose [2, 3]. A dose conversion program named as "LumiCon (Luminescence Conversion program)" was developed to convert a mobile phone dose into a human body dose under the various event of radiation accidents. Postures and size of a body, types and directions of an isotope, and mobile phone locations were considered as main parameters for large-scale radiation accident scenarios. Three postures (standing, kneeling, and squatting) were used with standard MRCP (mesh reference computational phantom) [4]. In addition, male and female phantoms with 10th, 50th, and 90th height and weight percentile were applied [5]. Three radiation sources (Ir-192, Cs-137, and Co-60) are selected with six exposure geometries (anterior-posterior (AP), posterior-anterior (PA), left-lateral (LLAT), right-lateral (RLAT), isotropic (ISO), and rotational (ROT)). Mobile phones on the representing positions such as chest, hip, thigh, and hand on the phantom were simulated using GEANT4 code. A display glass, resistors, and SIM card, which are well studied for retrospective dosimetry, were specially designed in a mobile phone structure as luminescence materials. A human body dose can be represented with five organ doses (red bone marrow (RBM), brain, lungs, small and large intestine) that might cause acute radiation syndrome or whole body absorbed dose, or W_T(Tissue waiting factor)-weighted body dose. The program can also provide a dose range regarding with un-known factors if a victim has limited memories (e.g. phone positions, postures). Moreover, clinical symptoms and emergency treatment regarding the amount of an exposure dose can be provided together with dose reconstructions.

Keywords: *emergency dosimetry, LumiCon (Luminescence Conversion program), thermoluminescence (TL), optically stimulated luminescence (OSL), mobile phone, mesh phantom, Monte Carlo simulation*

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PS2 (T2.4-0911)

Retrospective cytogenetic biodosimetry for Chernobyl clean-up workers

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Information about external radiation dose is one of the significant factors, impacting the decision about providing targeted medical care and prediction of medical radiological effects in individuals, which have been exposed to radiation. It is well know that majority of Chernobyl clean-up workers have been exposed to low doses of irradiation, however, many of them have no established external dose. Therefore there is a need to restore the information about radiation doses in the group of Chernobyl clean-up workers. A great number of studies have demonstrated the ability of biological cytogenetic dosimetry to estimate radiation doses not only soon after the exposure to ionizing radiation, but also retrospectively. Therefore the aim of the study was to evaluate the ability of cytogenetic analysis to estimate radiation doses retrospectively for Chernobyl clean-up workers though the potential of retrospective cytogenetic dosimetry to estimate low doses is restricted to the borderline of sensibility which is reported to be 10 cGy.

Cytogenetic assay has been made in 108 Chernobyl clean-up workers 27–30 years after their participation in recovery works. Stable and unstable chromosome aberrations were assessed in peripheral blood lymphocytes. Analysis of stable chromosome aberrations (FISH-analysis of translocations) was performed in 75 patients, analysis of unstable chromosome aberrations– in 74 patients. Analysis of unstable chromosome aberrations has revealed increased level of cytogenetic radiation markers in 45.5% of clean-up workers up to 30 years after the participation in accident recovery works. Biological radiation dose was determined in 18% workers using FISH-analysis of translocations and doses ranged from 14 to 48 cGy. Analysis of stable chromosome aberrations demonstrated the ability to estimate biological doses in a remote period after exposure to low-dose ionizing radiation. Besides, a number of factors (health state, oncology diseases, radiotherapy, chemotherapy) should be considered while evaluating radiation doses retrospectively in order to interpret the obtained results correctly.

Keywords: *Chernobyl clean-up workers, low doses, cytogenetic dosimetry.*

PS2 (T2.5-0180)

Computation of Excess Lifetime Cancer Risk for Environmental Exposures

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The research investigated the Excess Lifetime Cancer Risk (ELCR) of Natural radionuclide in soil around a Cement factory in kogi state, Nigeria. The experimental aspect of the study involved the determination of primordial radionuclides concentration in the soil sample using high efficiency gamma-spectrometry. The mean activity concentration of (^{226}Ra, ^{232}Th and ^{40}K) in the soil were, (47.2956, 77.5699, 601.8127) BqKg^{-1}, respectively. Transfer Factor (TF) of natural radionuclide from soil to selected cultivated crops around the farms were below the World standard of unity and varies in order of ^{232}Th > ^{226}Ra > ^{40}K. There exists strong positive correlation ($p < 0.05$) relationship between ^{226}Ra$_{soil}$ and ^{226}Ra$_{food}$ ($r = 0.9699$), The positive correlation found among the ^{226}Ra radionuclides between the soil and food samples suggest that these radionuclides originated from the same source for individual environmental matrix. Theoretical aspect involved the computation of AED and ELCR, with the aid of a computational code (Radbell, 001). The mean annual effective dose ranged from (0.557 – 1.549) mSvY^{-1}. Furthermore, the Excess Lifetime Cancer Risk ranged from (2.018 to 5.442) x10^{-3} with mean value of 3.45x10^{-3}. The mean value was slightly, above the world standard of 0.29 x10^{-3}. The slightly high level of Excess Lifetime Cancer Risk might be associated to high cement production activities associated with the area, without taking into consideration the radiological implication of the inhabitant.

Keywords: ELCR, AED, TF, Radionuclide and Computation.

PS2 (T2.5-0186)

Geochemical Assessment and Spatial Distribution of Heavy Metals in Food Crops and Soils of Agricultural Farmlands in the South-Western Region of Nigeria

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In this study, seven heavy metals such as Cu, Mn, Pb, Cr, Zn, Co and Cd have been examined in 500 food crops and 500 soil samples at corresponding locations from agricultural farmlands across the south western region of Nigeria. Concentrations of the heavy metals were determined using atomic absorption spectrometric (AAS) analysis and there health risk was determined. In order to assess the degree of heavy metal contamination in the food crops and soil, a geochemical assessment was calculated using the enrichment factor, the geo-accumulation index, pollution index and pollution load index based on the model released by the U.S. Environmental Protection Agency. From the analysis, traces of Co and Cd were detected in the farmlands, while the concentrations of Mn, Zn, Cu, Cr and Pb increased in that order at the unpolluted level. The concentrations of these metals were within Food and Agricultural Organization/World Health Organization (FAO/WHO) safe limit. Spatial distribution of the concentrations of the heavy metals was carried out using the GIS approach.

Keywords: Heavy Metal Contamination, Agricultural Farmlands, Geochemical Assessment, Spatial Distribution

PS2 (T2.5-0247)

Radiological risks assessment and the mapping of background radioactivity in groundwater of a basement complex area of southwest Nigeria

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The presence of primordial radionuclides in the ^{238}U and ^{232}Th decay series and non-series ^{40}K in groundwater is an important pathway of man's exposure to radiations in the environment. With increasing reported cases of lung cancer in Nigeria over the last decade, this study was initiated to carry out a comprehensive radiological risk assessment of natural radioactivity in groundwater and to generate a baseline background radiation map for the basement complex geological areas of southwest Nigeria. To accomplish these, samples of groundwater were collected from 100 locations within the different geological settings in the study area to assay for ^{226}Ra, ^{232}Th, ^{40}K and ^{222}Rn using gamma spectroscopy and continuous radon monitoring approach. Results show that ^{226}Ra, ^{232}Th, ^{40}K and ^{222}Rn present mean activity concentration values of 4.1, 3.4, 60.2 and 32.3 Bq/l, respectively. All these values are higher than the WHO recommended limits for drinking water. Generated maps identify areas of low, moderate and high radioactivity contents in groundwater.

PS2 (T2.5-0282)

Major ions and Uranium Content in Groundwater around some Large-scale Minerals Mining and Processing Sites in Nigeria

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In the absence of monitoring, impact assessment and remediation, the detrimental effects of industrialization in the environment may become unduly severe. Therefore, this study serves as an assessment of groundwater in the vicinity of some mineral mining and processing sites in Nigeria. The collected samples were analyzed for anions and cations concentrations as well as uranium-238 activity concentrations using analytical techniques such as ion-chromatography and Inductively Coupled Plasma - Optical Emission Spectrometry. Fluoride, sulphate and phosphate were found in some locations at concentrations that are unsafe for human consumption. The occurrence of these parameters above permissible limits alludes to the impact of anthropogenic activities in the investigated environmental media. However, the activity concentration of uranium-238 in the groundwater has a range of 0.00 to 47.50 ± 0.06 Bq/l and these values are below the recommended limit of 100 Bq/l as suggested by relevant local and international regulatory agencies. The results are expected to engender possible interventional actions in the areas of concern and provide useful data for the regulation of further industrial activities in the study locations.

Keywords: *Uranium, groundwater, Minerals mining and processing*

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PS2 (T2.5-0292)

Cyclotrons – Radiological impact assessment

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Cyclotrons are particles accelerator used to product radionuclides for medical application like diagnostic and therapy mainly in cardiology or oncology.

Radionuclides are produced in cyclotron bunker and then transferred into radiosynthesis units where they are coupled with pharmaceuticals. Radionuclides produced are Beta plus emitters and short half-lives as for example F-18, C-11 and O-15. Most of the production units are located in urban area.

Assessment carried out by the French institute for radiation protection and nuclear safety (IRSN) feedback pointed out difficulties linked to the following findings:

- A misuse of Gaussian model for atmospheric dispersion due to the location of production units (short distances between person of interest and the point of discharges.
- A range of dose estimation which might be higher than nuclear power plan's.

The Institute adapted its methodology for radiological impact assessment to take into account the sites specifies particularly on atmospheric dispersion model used and its qualification. For this purpose, targeted measurements are realized in the vicinity of production units. They concern:

- In situ measurement of F-18 into the atmosphere,
- Tracing dispersion mechanisms with stable helium,
- Acquisition of micro meteorological data
- Comparison between measurements and modeled calculation.

The aim of this presentation is to describe the methodology used by the IRSN for radiological impact assessment of radiopharmaceuticals production units and previous results of measurements campaign.

Keywords*: cyclotron, impact assessment*

PS2 (T2.5-0298)

The thoron issue: exhalation rate and interference with radon measurements performed with passive radon dosemeters

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It is universally recognized that very seldom thoron (^{220}Rn $t_{1/2}$=55.8 seconds) can be considered as a significant source of exposure to population and workers. Actually, its short half life limits in a substantial way its capability to reach high indoor concentrations in dwellings and workplaces. In typical indoor environments, thoron concentrations are at least one order of magnitude lower than those of radon, the most long lived radioisotope (^{222}Rn $t_{1/2}$=3.82 days) and for that reason only rarely the doses delivered to humans by thoron is really important. A significant exposure to thoron may arise only in very special cases, for example when bricks containing high ^{232}Th levels are used as main material in building constructions.

However the presence of thoron can't be simply disregarded even in standard situations, i.e. when the concentrations of the thorium series radioisotopes in soil and building materials are similar to those of the uranium series ones. Actually, it can be demonstrated that the thoron exhalation rate from the walls of a room can influence the radon concentration measurements performed by means of passive radon detectors: the thoron concentrations close to the walls of a room are in fact normally much higher than those in the center and because of that the dosemeters may be exposed to quite high thoron concentrations. In these cases the radon measurements performed with passive devices may be significantly flawn.

It is therefore important: 1) to assess the response to thoron of the radon measurements passive devices; 2) to evaluate the thoron exhalation rates of the different building materials. In this work these two issues are explored and discussed in details by means of experimental and theoretical considerations. In particular some experimental data showing the thoron influence on some typical passive radon detectors are presented and a simple method for the evaluation of the thoron exhalation rate from building materials is proposed.

Keywords*: Thoron, Exhalation rate, Passive detectors*

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Poster Session T2

PS2 (T2.5-0299)

Assessment of Radon Contribution from Tailings Dams

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Air dispersion models have been used to determine the origin of radon source and to predict the impact of the gold mine tailings as radon source on the air quality. However the accuracy and reliability of this approach is severely hampered by the assumptions and simplifications that have to be made. This study developed a technique to validate radon dispersion modelling that assesses the radon contribution from typical tailings dams.

Industrial Source Code Source Term 3 (ISCST3) Gaussian model simulations on ^{222}Rn dispersions were computed over five days. Dispersion modelling scenarios included four different tailings geometry sources (true geometry, flat-ground level area, top-level area, volume source) and the wake effect. Model evaluation and correlation of model predictions with field measurements were computed using five standard model validation statistics. The "age" of the gas approach was used to apportion the source, thereby validating the model.

Model results highlights the effects of source geometry and wake effects at near-field ground level receptors. The highest concentration predicted by the model from the true geometry source was found to be 0.843 Bq/m^3, which correspond to the dose of 0.012 mSv/y to the public of due to radon from the tailings. This value is less than the 1 mSv/y dose constraint stipulated by National Nuclear Regulator. The results further showed a constant trend for all the modelling scenarios and the model performance was particularly acceptable for true geometry area and flat ground area sources. The model was successfully validated by tracing the origin of radon source back to the tailings as predicted by the model.

PS2 (T2.5-0358)

"Novel platform for detection of contaminants in the environment with high reliable spectroscopic measurements"

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Abstract: Increased sensitivity to nuclear safety and security has prompted public entities and private institutions to maximize their capability to rapidly assess risks and intervene in the case of accident or threat. Quick intervention and response are achieved through nuclear measurements via airborne, land, and underwater systems. A network of cohesive, well-integrated and easy deployable radiation monitoring systems combined with real-time analysis of data is essential to facilitate and enhance the decision-making process during these most critical moments enhancing the quality of the management plan.

The presented radiation monitoring systems can be integrated in several form factors which depend mainly on operational needs and internal battery for autonomous operation. Compact ARM based computers is embedded, which can store large amount of data in their non-volatile memory, run automatic data analysis and trigger alarms in case of exceeding radiation levels. All the systems can communicate with redundant interfaces in failover configuration and upload the acquired environmental information in a central database. The same monitoring systems can alert the emergency response personnel on the field as well, through wireless connection to common tablets or cellphone or SMS, guaranteeing a prompt response in case an illicit transportation of radiological or nuclear material is detected.

PS2 (T2.5-0423)

Routine Monitoring around Belarusian NPP for Radiation Protection of the Public

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Republic of Belarus is constructing its first NPP under the Russian design AES-2006, consisting of two nuclear power units of WWR reactors with power output to about 1200 MWe (gross) and located in the Ostrovets area Grodno region. Unit 1 is planned to commence operation by the end 2019.

According to the IAEA recommendations and Belarusian national regulation the monitoring programme for nuclear facilities such as NPPs should be developed and realized at all stages in the lifetime of the nuclear facility. A regular monitoring allows to ensure the protection of the public and the environment.

By 2019, pre-operational studies were performed for Belarusian NPP to establish 'baseline' environmental radiation levels and activity concentrations for subsequently determining the impacts of this facility.

Radiation monitoring around the NPP in Belarus is performed by the operator of the NPP, by the Ministry of Environment and by the Ministry of Health. The Routine radiation monitoring is carried out in the observation zone of Belarusian NPP (12.9 km radius around the site with the population of about 8 thousand people) and in the Ostrovets city (the nearest to the site big settlement) scince 2016. The constituents monitored are: activity concentrations is air, atmospheric fallout, surface water, freshwater seaweed, freshwater fish, sediments, drinking water, soil, some types of plants, local food products (like milk, meat, grain, potato, leafy and root vegetables, tec.), forest products (mushrooms and berries). The Ministry of Health, as one of the regulators, is responsible for dose assessment for public and collects the data about activity concentrations in foodstuff and drinking water and about parameters needed for dose assessment purposes (such as food consumption rates, habit data, etc.).

In order to limit radiation exposure of the public from Belarusian NPP a dose constraint of 100 µSv per year (50 µSv for gaseous releases and 50 µSv for liquid discharges) was established in national regulatory documents, as well as an exemption level of 10 µSv per year. Atmospheric and liquid discharge limits were specified for the particular radionuclides and groups of radionuclides on the basis of their special radiological significance.

Prognosis estimates of total effective dose to public in case of normal operation of Belarusian NPP were made using the collected data about population distribution, lifestyles, habit data, consumption rates data, etc. typical for the region as well as data about prognostic values of radioactive atmospheric releases and liquid discharges. Doses to public in case of normal operation of the Belarusian NPP are expected to be about 0.2 µSv/year at 1 km from the site for atmospheric releases and about 8 µSv/year at 1.5 km from the discharge point – for liquid discharges. These values do not exceed established dose constraints.

The results of routine monitoring at the operational stage will be used for providing information about the impact of Belarusian NPP on public health and the environment to stakeholders in the country and international communities.

Keywords: Belarusian NPP, radiation monitoring, public protection

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PS2 (T2.5-0499)

In-situ gamma-ray spectrometry method for rapid investigation of radioactive contamination of rare earth minerals

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An in-situ gamma-ray spectrometry survey with a scintillation detector of NaI(Tl) (φ75 mm × 75 mm) was carried out in Baotou City and Bayan Obo Districts in order to estimate the levels of natural radionuclides near rare-earth (RE) tailings dams. In the RE tailings dam of Baotou, the mean concentrations of ^{238}U and ^{232}Th were 3.0 ± 1.0 mg/kg , ranged from 1.9 to 4.6 mg/kg, and 321 ± 31 mg/kg, ranged from 294 to 355 mg/kg, respectively. In the Bayan Obo tailings dam, the mean concentrations of ^{238}U and ^{232}Th were 5.7 ± 0.5 mg/kg (range: 5.3 - 6.1 mg/kg) and 276 ± 0.5 mg/kg (range: 275.5 - 276.3 mg/kg), respectively. The average ^{232}Th concentrations in the mining areas of the Bayan Obo Mine and the living areas of the Bayan Obo Town were 18.7 ± 7.5 and 26.2 ± 9.1 mg/kg, respectively. The ^{232}Th concentration recorded in the tailings dams was much higher than the global average (7.44 mg/kg). Our investigation shows that the ^{232}Th concentration in the tailings in the Baotou dam was 34.6 times greater than that in the local soil (in Guyang County); the average concentrations of ^{232}Th in the soil in the Baotou District and Bayan Obo Districts were about 1.35 and 2.82 times greater, respectively, than that in the soil in Guyang County. The highest estimated effective dose due to gamma irradiation was 1.15 mSv per year, estimated from the data observed in the Baotou tailings dams. This result is basically consistent with the annual effective dose calculated by the measured results of the dose rate meter. The results of this preliminary study indicate that the RE tailings dams may be potential contamination sources of radioactivity and that remedial measures may be required.

Keywords: *in-situ gamma-ray spectrometry, rare earth minerals, tailings dams*

ACKNOWLEDGMENTS
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PS2 (T2.5-0520)

Releases of Medically Used I-131 into the Environment

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Among the radionuclides used in nuclear medicine, radioiodine (I-131) can enter the environment to a measurable amount due to its half-life of 8 days and the frequency of its application.

Generally, the radiation protection measures in Germany, as given in a guideline of the German Environmental Ministry (Ref. 1.), result in retention of the major amount of I-131-activity applied in nuclear medical hospitals within the hospitals. Therefore, the hospitals themselves are not a relevant source of I-131 into the environment.

An important pathway of I-131 into the environment is from the patient's excreta to the wastewater in wastewater treatment plants (WTP). Another path of propagation leads via contaminated sanitary products into household waste to waste incineration plants (WIP). From WTP a part of the I-131 is released into rivers, but another part remains in the sewage sludge. The incineration of sewage sludge and household waste in WIP can release I-131 into the atmosphere.

On the basis of measurements of I-131 in sewage treatment plants and the evaluation of results obtained at inspections of waste regarding orphan radiation sources, the I-131 activities released into the environment in Germany are estimated. It is shown that the release of I-131 via household waste is significantly lower than via wastewater.

Keywords: *Radioiodine, wastewater, atmosphere*

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PS2 (T2.5-0605)

On-line gross α/β measurement for tap-water monitoing:a preliminary study

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Abstract: Low-level radioactive contamination measurement in water requires low-background, high-efficiency and large-area detector. In this work, we present the construction and performance of a measurement instrument with the aim to on-line monitoring the gross α/β radioactivity concentration in tap-water and to react or alarm instantly when the radioactivity concentration is above the detected radiological threat.

This on-line monitoring instrument consists of detectors, inlet waterway and outlet waterway, data acquisition (DAQ) circuit, computer software program. EJ-444 detector with a surface area 457mm×457mm consisting of a EJ-212 plastic scintillator coated with EJ-440, a polycrystalline power containing the ZnS:Ag is used to meanwhile detect α/β radioactivity. In order to increase the effective detection area and improve the detection efficiency, several EJ-444 detector foils are used and the light generated by the detector is transmitted through hundreds of wave length shift(WLS) fibers which are embedded in the PMMA between two detector foils. Inlet waterway and outlet waterway include filter, solenoid valves and pumps which are used to adjust the balance of the water inlet and outlet ensuring continuous on-line measurement. Data acquisition (DAQ) circuit is intended to digitalize the pulse waveform immediately and to store the data in the form of signal amplitude and time using ADC chip with sampling rate above 1Gsps. With the aim to facilitate the system operation and realize the digital signal processing (DSP) and visualization, computer software program can match well with the DAQ circuit interfaces and accomplish the functions of data inquiry and information management based on database.

The following instrument parameters and measurement preliminary results are tested: characterization of EJ444 detector and output signal, background, short/long term stability, selectively discrimination of α/β ray using radioactive sources.

In order to reduce the background level brought by the surroundings, 316L stainless steel with 1cm thickness is used as the frame structure of the instrument.

Key words: gross α/β monitoring; EJ444 detector; α/β discrimination

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PS2 (T2.5-0615)

Evaluation of Minimum Detectable Activity for Underwater Radiation Monitoring System

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Water is one of the most important natural resources and has to be prevented from any contaminations. In particular, small amounts of artificial radioactive materials with long half-life such as ^{137}Cs which can be generated by an accident of nuclear facilities could cause radioactive contamination in water, extensively. The interest in the radioactive contamination issues have been increasing all over the world since the Fukushima nuclear facility disaster in Japan, and various radiation monitoring system has been developed to monitor radioactive concentration in air. However, it is still a challenge to monitor radiation in water due to its detection distance, supply of electric power, water-proof design, and so on. Recently, researchers at the Korea Atomic Energy Research Institute (KAERI) have developed an underwater radiation monitoring system to detect ^{137}Cs (662 keV) and ^{131}I (364 keV), which are produced by nuclear fission. The monitoring system is designed to float on the water by buoy and use solar panel for self-powered operation. In this study, various experiments and simulations were carried out to determine a detector sensor taking the cost to benefit ration into account and evaluate the minimum detectable activity and the activity conversion factor to operate the system.
There are many types of detector sensor and they have their own properties. Among them, BGO, CsI(Tl), $LaBr_3$(Ce) and NaI(Tl) cylindrical radiation detectors (2 inch in diameter and 2 inch in length) were considered for the system. Their performances were estimated by a Monte Carlo simulation in terms of the energy resolution and detection efficiency and compared each other. The BGO detector provided the best performance, the NaI(Tl) detector, however, was chosen taking the cost to benefit ratio into consideration. Based on the results of this, an NaI(Tl) detector was associated with multi-channel analyser (MCA) in a waterproof device. To measure the activity conversion factor, a tank (1.9 m in diameter and height) was filled with water and ^{68}Ga radioactive source was mixed in water instead ^{137}Cs and ^{131}I due to its long half-life which could cause radioactive contamination after experiments. The detection signals along with the source activity were measured and compared to the simulation data under the same conditions. Based on the comparison, the activity conversion factors for ^{137}Cs and ^{131}I were estimated to be 1.74×10^{-2} (Bq/L)/count and 1.30×10^{-2} (Bq/L)/count, respectively. The minimum detectable activity was calculated from the background detection data in water and the activity conversion factor to 0.77 Bq/L for ^{137}Cs and 0.89 Bq/L for ^{131}I.

Keywords: *Underwater Radiation, Radiation monitoring, Radioactive Contamination*

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PS2 (T2.5-0631)

Development of Radiation Early Warning System for Underwater Operation in the Marine Environment

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Since the Fukushima nuclear disaster, there have been growing interest worldwide to monitor its marine environment by periodical sampling and/or installation of unmanned detection system. Development of unmanned radiation detection system for marine underwater environment, however, requires the instrumentation of detection equipment and the necessary software design for data exchange and data analysis. In this study, various designs of the current underwater radiation monitoring systems are introduced and our methodologies to optimize the radiation detection functionality in terms of instrumentation are shown.
The system is composed of the detection module based on NaI(Tl) scintillator with multichannel analyzer (MCA), the water chamber specially designed in accordance with the result of MCNP simulation and the local controller unit for data transmission of real time underwater radioactivity.

Keywords: Radiation, Early Warning, Underwater, Marine, Fukushima

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PS2 (T2.5-0639)

In-situ gamma measurement within 8 MeV in Jinping underground laboratory

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The China Jinping underground laboratory (CJPL) is the deepest underground laboratory in the world with the rock overburden 2400 meter. Some dark matter experiments, neutrinoless double beta decay experiments and nuclear astrophysics would be carried on CJPL-II. The gamma spectrum within 8 MeV in CJPL is very important for nuclear astrophysics experiment design and running. A NaI(Tl) detector with Φ5inch*5inch was used to in-situ measure in CJPL-II. To compare the gamma spectrum in different environment, the NaI(Tl) detector was measured in Tsinghua campus above and Jinping underground with/without lead shielding. with this results, the low energy background within 3 MeV of CJPL is closed to that of Tsinghua campus. the High energy background within 3-8 MeV of CJPL is one or two orders of magnitude lower than that of Tsinghua campus. The details would be introduced in this paper.

Keywords: *CJPL; sodium iodide detector; high energy gamma background*
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PS2 (T2.5-0640)

Radiological Risk Assessment for VVR-S Research Reactor Radioactive Effluents Underground Buffer Tank Dismantling

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The VVR-S nuclear research reactor from "Horia Hulubei" National Institute of Physics and Nuclear Engineering (IFIN-HH), Bucharest, Romania was operated between 1957 and 1997. The main purpose of reactor was the radioisotope production in the thermal column for medical and industrial purposes as well as other research activities. Based on the preliminary radiological characterization the selected decommissioning strategy, consisting in immediate dismantling of the contaminated and activated components and structures, was carried out in between 2010 – 2020. The reactor was provided with a system for collecting radioactive leaks consisting of radioactive liquids resulted from hot cells decontamination, showers and other spills coming from equipment's of the primary circuit (pumps, de-aerator, filters, etc.). The main component of this system was the 30 m3 underground buffer tank for radioactive liquids intermediate storage, berried in soil at a depth of 5.5 – 6.5 m and located in the immediate vicinity of the reactor. The radioactive liquids were transferred from the reactor components and structures to the buffer tank and further to the Radioactive Wastes Treatment Plant using an underground stainless-steel pipes system. As part of the decommissioning process, the dismantling of the buffer was performed in 2018 using an ALARA methodology. The evaluation of the radiological risks for the workers involved in the buffer tank dismantling was performed using RESRAD-Build code and for the environmental radiological effects of this operation the RESRAD-ONSITE Code was used. The modelling results were compared with real in-situ dose measurements for the workers and the activity concentrations of the soil and water samples from the dismantling area were compared with the estimated values.

***Keywords**: Buffer Tank Reactor Decommissioning, Workers and environmental risk assessment*

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PS2 (T2.5-0680)

Discrimination of Abnormal Environmental Dose Rate Data by K-means Clustering Algorithm

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Environment dose rate is determined by the intensity of cosmic rays as well as kinds, concentration and distribution of radioactive nuclides, which are affected by or associated with a lot of factors. The relationship of them is hard to determine because of its complexity. Machine learning algorithm is used to work out the problem in this paper. First, we generate series of data based on environmental monitoring, which include time, site, dose rate, temperature, wind speed, wind direction, humidity, atmosphere pressure and precipitation. And we study day, season, year variation of these data. The result shows that huge dose rate variation is mainly associated with precipitation and humidity. Then we use K-means Clustering Algorithm, an unsupervised algorithm, to divide the data into 3 categories: normal working condition, variation caused by precipitation and variation caused by other circumstances. Based on this, we are going to determine which category a data is belonged to.

Keywords: *environmental monitoring, K-means Clustering Algorithm, dose rate*

Poster Session T2

PS2 (T2.5-0690)

Korean Data on Radionuclide Transfer to Rice Plants and Proposal for Its Use in Radiological Impact Assessment

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Rice is the most important staple food in many Asian countries including Korea and Japan. After the Fukushima DNPP accident in 2011, many Japanese people were worried lest the radioactivity in rice should be above the safety level. The Fukushima DNPP accident occurred approximately two to three months before rice transplanting. As a result, root uptake of radionuclides from contaminated soil, referred to as indirect transfer pathway, was of great concern. If the accident had occurred during the rice-growing season, direct activity deposition onto the rice plants might have received much more attention than the indirect pathway. In Korea, the direct and indirect pathways of some selected radionuclides in rice plants have been studied through greenhouse experiments by KAERI since 1981.

For studying the direct transfer pathway, a solution containing ^{54}Mn, ^{57}Co, ^{85}Sr, ^{103}Ru, and ^{134}Cs was sprayed over a canopy of rice plants grown in large lysimeters at six different growth stages. Three hours after the spraying, half the plants in the lysimeters were removed for quantifying the initial plant deposition. The rest were removed at normal harvest time for measuring the remaining activities. The transfer parameters investigated were the interception fraction (IF), weathering half-life, and seed-translocation factor (TLF). The highest IF value measured was around 0.94 for all of the radionuclides. In the TLF values, differences by factors of up to several tens to hundreds were found between the most mobile radionuclide (^{134}Cs) and the most immobile ones (^{103}Ru and ^{85}Sr). Experiments on the exposure of rice plants to HTO and I_2 vapor at different growth stages were also conducted.

Regarding the indirect transfer pathway, the traditional soil-to-plant transfer factor (TF, dimensionless) and the areal transfer factor (ATF, m^2 kg^{-1}) have been studied. The ATF can cover any pattern of the activity distribution in soil, provided that data on the initial activity deposition or present inventory is available. To produce TF data, various soils were labeled with ^{54}Mn, ^{60}Co, ^{65}Zn, ^{85,90}Sr, ^{99}Tc, ^{125}I, and/or ^{137}Cs for the upper 15~20 cm soil layers of pots or small lysimeters sometime before rice transplanting. TF values were also measured for fallout ^{137}Cs in several paddy fields. ATF values of ^{54}Mn, ^{60}Co, ^{85,90}Sr, ^{99}Tc, ^{125}I, ^{137}Cs, and/or HTO in various soils were measured for their deposition onto the surface water in lysimeters at different growth stages.

Considerable variations in the parameter values were generally found among the soils, radionuclides, and/or deposition times. Most of the produced data are applicable to the radiological impact assessment for both the normal chronic and acute accidental depositions. For the former, it may be necessary to determine the single representative values of such parameters as IF and TLF based on the deposition time-dependent data for each radionuclide. The TF data are useful for setting up reference levels of paddy soil contamination as a standard for deciding the banning of rice culture. A great portion of the parameter data was included in the TECDOC-1616 and TRS-472 of the IAEA, which were published through the IAEA's EMRAS program.

PS2 (T2.5-0726)

Investigation and assessment of environmental gamma radiation dose rate surrounding area of Haiyang Nuclear Power Plant, China

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Abstract: Units 1 and 2 of Haiyang Nuclear Power Plant in Shandong province, China have been connected to the grid for power generation in August and October, 2018, respectively. In order to obtain the background data of environmental gamma radiation dose rate in the surrounding areas before the operation of Haiyang Nuclear Power Plant, and provide basis for evaluating the impact on the surrounding environment after the operation of the nuclear power plant. The environmental gamma radiation dose rate was measured by vehicle radiation monitoring system (GR460) in June 2018 and the resultant radiation dose exposure to local residents was also estimated. The air absorbed dose rate ranges from 39.6 to 109 nGy/h, and the average is 72.2 nGy/h. There is a significant difference between the mean values of gamma radiated air absorbed dose rates in the regions of 0-5 km, 5-10 km, 10-20 km and 20-30 km of nuclear power plant. The per capita annual effective dose of residents caused by gamma radiation dose rate in outdoor environment is 84.8 μSv. The environmental gamma radiation dose rate in the surrounding areas of Haiyang Nuclear Power Plant and the resultant radiation dose exposure to the local residents was within the normal background level in China.

PS2 (T2.5-0729)

Performance of a Sr Adsorbent for Sr Monitoring in Water Sample

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Introduction

Monitoring of environmental radiostrontium isotopes, ^{89}Sr and ^{90}Sr, is important because radiostrontium is known to accumulate in bone and to increase risks of bone cancer and leukemia. Since they emit no γ-rays but solely emit β-rays, separation from other radionuclides before measurement is indispensable. The conventional methods to measure radiostrontium are time-consuming complicated procedures using lots of deleterious substances. We are investigating simple methods to analyse radiostrontium in environmental samples. There is a new Sr adsorbent; 'Pureceram MAq®' (Nippon Chemical Industrial, Co. Ltd.) which has produced in order to adsorb Sr in water. The Pureceram MAq is composed of barium silicate and is not water-soluble, and the particle size is from several μm to few hundreds μm. We tried to use the adsorbent for measurement radiostrontium concentration in water. In this study, the property of the adsorbent was experimentally analysed and the feasibility of the adsorbent to use the measurement of radiostrontium in water was considered.

Materials and Methods

(1) Confirmation of required amount of Sr Adsorbent

The adsorbent, each amount was 20, 60, and 100 mg, was added to 100-mL artificial seawater spiked with ^{90}Sr. The seawater was stirred for 60 min. Finally, the water was filtered with a filter (pore size 0.45 μm). The ^{90}Sr activity on the filter was measured with a low background liquid scintillation counter (LSC LB-7, Hitachi, Ltd.) and that in the filtrate water was measured with a liquid scintillation counter (LSC-7400, Hitachi, Ltd.).

(2) Change-with-time of Sr concentration

The adsorbent of the amount 100 mg was added to 100-mL artificial seawater spiked with ^{85}Sr. The seawater was stirred and the fractions were subtracted along the stirring. The fractions were filtered and the ^{85}Sr activity was measured with a gamma counter (ARC-7001, Hitachi, Ltd.). The ^{85}Sr activity on the filters was measured with a HPGe detector.

(3) Adsorptivity of Na, K, Mg, Ca, and Cs

Adsorptivity of Na, K, Mg, and Ca were analysed with an ICP-AES (Thermo Jarrel Ash, IRIS/AP), and that of Cs was analysed using ^{134}Cs.

(4) Application to inland water

The strontium adsorbent was applied to adsorb Sr in inland water.

Results and Discussion

It was found that 100 mg of the adsorbent is required to adsorb Sr in 100-mL seawater. It was verified that 120 min stirring is enough to adsorb Sr in seawater. The concentrations of the elements, Na, K, Mg, Ca, and Cs in the liquid phase were constant throughout the stirring. Thus, it was proved that the adsorbent adsorbs Sr selectively. It was confirmed that the adsorbent adsorbs Sr in inland water when there exists sulfate ion.

Solely certain time of stirring is sufficient to adsorb Sr with Pureceram MAq without any other chemical procedure. The adsorbent may be applicable to use radiostrontium monitoring of seawater and inland water. However, one should take care the self-absorption of beta-rays emitted from ^{90}Sr and/or ^{89}Sr. When the adsorbent is used for radiostrontium monitoring, one should pay compensation of the self-absorption.

Conclusion

It was confirmed that the Sr Adsorbent selectively adsorbs Sr in seawater and in inland water. Feasibility of monitoring Sr in water sample using the adsorbent was indicated.

Keywords: *Radiostrontium, Monitoring, Strontium adsorbent*

PS2 (T2.5-0772)

Airborne Be-7 and artificial Cs-137 radionuclides in mosses of Armenia

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Activities of air-migrated Cs-137 and cosmogenic Be-7 radionuclides were investigated in the moss samples collected throughout Armenia in the period 2016-2019. The sampling campaign was implemented within ICP Vegetation moss survey (35 samples) and Radioecological monitoring in the area of Armenia – REMA II project (48 samples). The most widespread moss species (*Bryum argenteum, Homalothecium philippeanum, Ptilium crista-castrensis, Syntrichia ruralis*) were selected at each sampling location. At the laboratory mosses were air-dried, manually cleaned and analyzed on a low-background gamma-ray spectrometry system (HPGe detector coupled to multichannel analyser, Genie 2000 software, CANBERRA). The counting time was 30000 sec. After background correction the activity concentrations (Bq/g) of Be-7 and Cs-137 were determined in Genie 2000 software from energy peaks of 477.60 and 661.5 keV, respectively.

Although no significant correlation (Spearman non-linear correlation) between the activity of radionuclides and altitude was found for the overall dataset, such correlation was observed for samples of separate mountain ridges and massifs. Be-7 activity ranges from <MDA to 0.283 Bq/g (n=83, valid N=38) with the mean value of 0.071 Bq/g. The highest activity for Be-7 was observed in the moss sample from the area of Aragats Mountain (from 3200 m), located in the central part of the country. For artificial Cs-137 the activity ranges were <MDA to 0.297 Bq/g (n=83, valid N=46) with the 0.064 Bq/g mean value (Fig. 1). The highest activity was observed in the moss from 2600 m a.s.l., again in the area of Aragats Mountain. The distribution of artificial Cs-137 and natural Be-7 in mosses enables the development of a Radioecological database for Armenia.

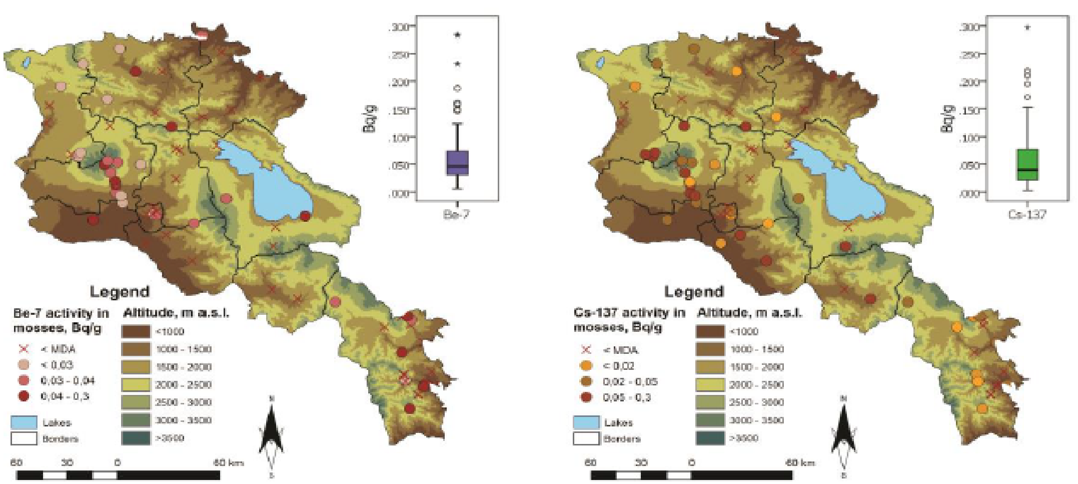

Fig.1. Distribution of Be-7 and Cs-137 radionuclides in mosses of Armenia by altitude

Keywords: Cs-137, Be-7, Moss, radionuclides

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PS2 (T2.5-0861)

Continuous Monitoring of Temporal Variation in Background Gamma Dose Rates and the Potential for a National Environmental and Radiological Monitoring Network in Nigeria

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International As part of the developments within the proposed national environmental and radiological monitoring program in Nigeria, a continuous gamma dose-rate monitoring device (Gamma Tracer (GT)) was installed on the Alabata Campus of the Federal University of Agriculture, Abeokuta, Nigeria. The experience and data obtained during the equipment pre-installation performance characterization, as well as the monitoring data collected after its installation are presented in this paper. Analysis of the preliminary GT data during the transit from the port city of Lagos in the coastal sedimentary area to Abeokuta, which is underlain by basement rock showed correlation between gamma dose-rates and geological formation and confirmed that Abeokuta is one of the high background radiation areas in Nigeria. The temporal dose rate values also showed seasonal variation and influence of meteorological parameters, such as air temperature, rainfall, relative humidity, etc. Statistical analyses showed that these meteorological parameters could account for up to 11% of the temporal variation in the gamma dose rate values. The implications of these findings as part of the potential technical challenges in environmental surveillance and monitoring for nuclear and radiological materials are discussed in this paper.

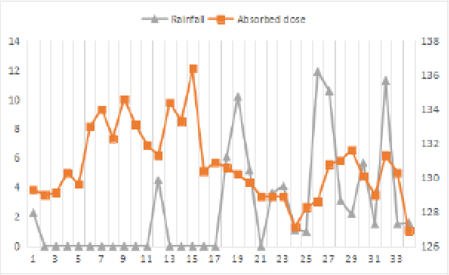

Fig. 1. Seasonal Variations (Dry and Wet) with Gamma Dose Rates

***Keywords**: Continuous, Temporal, Gamma*

ACKNOWLEDGMENTS

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PS2 (T2.5-0881)

Spectral Resolution Evaluation for Airborne Alpha and Beta Detection System According to the Shape of Collimator

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Alpha/Beta spectrometry is used to identify and quantify the alpha/beta radionuclides emitted in the decay process. The resolution enhancement is essential to analyze their spectrum due to the property of losing their energy during passing through the medium in air. [1] In order to develop an airborne alpha and beta detection system, the PIPS (Passivated Implanted Planar Silicon) detector is used to the property of high sensitivity. In addition, the collimator added to the detection system helps to improve the spectral resolution. In this paper, a hexagonal, circular, and pixelated collimators are used in order to confirm the change of resolution and detection efficiency. Figure 1 shows the cross section of the system according to the shape of collimator.

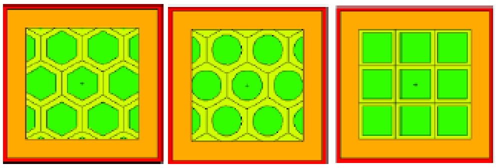

Fig. 1. Diagram of the detection system (Hexagonal, circular, and pixelated collimators in order)

The MCNP6 code is used for the simulation program. The Al collimator is placed between the detector and air filter. The radionuclides ^{218}Po(6.002 MeV), ^{214}Po(7.687 MeV) and ^{212}Po(8.784 MeV) are considered. The spectra are generated by using F8 (pulse-height) tally. The specifications of collimator are equivalent between three shapes of collimators, where the length of one side of the hole is 16 cm and the number of holes for each collimator is 9. Table 1 shows FWHM value and detection efficiency according to the shape of collimator.

Table 1. Spectral resolution and detection efficiency according to the shape of the collimator

	FWHM [MeV]	Efficiency (%)
Hexagonal	0.341977	0.011066
Circular	0.329077	0.010175
Pixelated	0.350613	0.013404

The circular collimator has the lowest FWHM, while the lowest efficiency. The efficiency was reduced by 32% and resolution was improved by 7% compared with pixelated one. Thus, the circular collimator was determined to be optimal because higher resolution is advantageous for radionuclide identification in alpha beta detection. It is possible to produce a high resolution and improve the spectral analysis by applying the circular collimator.

***Keywords**: Alpha and Beta Detection, Spectral Resolution, Detection Efficiency, Monte-Carlo Simulation*

ACKNOWLEDGMENTS

This work was supported by the 'Development of Portable Radioactive Contamination Monitoring System for Alpha and Beta Dust Source in the Air' of the Korea Institute of Energy Technology Evaluation and Planning (KETEP) granted financial resource from the Ministry of Trade, Industry & Energy, Republic of Korea (No. 20171510300590)

PS2 (T2.5-0897)

Meteorological effects on the performance of a solid-state α-detector measuring the water samples with predefined radon concentration

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The main aim of the study was to use α −spectrometry method to determine the functionality of a RAD-7 solid-state alpha detector. The known concentration of ^{226}Ra was measured in a RadioAnalytical laboratory. Different concentrations of ^{222}Rn were prepared. The RAD -7 detector was connected to the sample containing concentrations of ^{222}Rn as a daughter isotope of ^{226}Rn. The RAD-7 detector unit is equipped with an internal pump bubbling air though the water to release ^{222}Rn from the water sample. The bubbled air was dried up using the decadence and flown through the filter into the detector chamber of the RAD -7 where measurement of alpha particles occurred. The actual concentrations of ^{222}Rn using an indirect measurement of ^{222}Rn was achieved by measuring ^{218}Po and ^{214}Po daughters in secular equilibrium in a continuous enclosed system. The known standard samples concentration was compared with the measured standard samples results. A good correlation was observed with the range concentration from 90.66±7.20 mBq.l^{-1} to 314.65±24.6 mBq.l^{-1}. The two peaks with resolutions (FWHM) 19 keV (^{218}Po) and 21 keV (^{214}Po) shifted to the right by 50 keV and 210 keV, respectively and 166keV for LSC. The energy-tailing of the LCS spectrum was caused by Compton Scattering and led to higher count-rate of 4.11 cpm and 0.71 cpm for LSC and RAD-7 respectively. The meteorological increase of internal temperature and an increase relative humidity inside the detector resulted in a decrease ^{222}Rn concentration detected. An influence of external temperature to the water sample result in an increased ^{222}Rn concentration due to increase emanation of radon from water as a function of temperature increase. Also, an increase in sample volume, showed an increase ^{222}Rn concentration. The study demonstrated that RAD-7 is the suitable equipment for the measurement of radon concentration in water because of its sensitivity.

Keywords: : α −spectrometry, standard sample, LSC, radon,

PS2 (T2.5-0919)

Study of angular and energy response of various TLD holders for environmental dose measurements

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The presented study was made within the framework of EMPIR Preparedness project "Metrology for mobile detection of ionizing radiation following a nuclear or radiological incident". As a part of the work package for passive dosimetry working on improving methods for long-term monitoring in which one of the tasks was to investigate the mechanical characteristics of the dosimeter holder, which have an influence on the results of measured doses. [1]

Thermoluminescence dosimetry is a common method of personal and environmental monitoring. There are many thermoluminescence substances available. One of the most common are TLDs made from lithium fluoride with different admixtures. Moreover, various types of holders for the TL pellets are available. The combination of used TL material and holder may affect the results of the dosimeter.

The aim was to study the influence of holder type on the angular response in terms of ambient dose equivalent. Therefore in each holder the same TLD material was used. MCN-P pellets made of LiF:Mg,Cu,P were used in this study. Four commercial available holder types were tested together with one prototype made in CLOR. Three of them were based on basic RADOS holder (RADOS TLD Dosimeter for RE-2000 Readers manufactured by Mirion Technologies).

Irradiations were carried out in Central Laboratory for Radiological Protection (CLOR) in gamma calibration stands in the radiation beam generating by four gamma sources: ^{137}Cs, ^{60}Co, ^{226}Ra, and ^{241}Am. For each gamma source irradiations in at least 3 angular points were carried out (0°, 45° and 90°). For unsymmetrical holders additional 2 angular points of 135° and 180° were carried out. In each measurement point, 3-5 dosimeters were used. At each angle, the same conditions were preserved (a distance of reference point to the dosimeter from the source, irradiated dose).

Additionally, besides the irradiations, the Monte Carlo simulations were carried out.

Keywords: *TLD, angular response, Monte Carlo*

ACKNOWLEDGMENTS
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PS2 (T2.5-0920)

Radioactivity Evaluation Considering the Depth Profile Function

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In the event of a nuclear or radiological emergency, it is necessary to characterize contaminated area and to evaluate the radioactivity because of performing dose and risk assessment. In-situ gamma spectrometry can provide a rapid and simple method to measure the radioactivity of contaminated area. In order to estimate the radioactivity of soil, the distribution of radionuclide have to be taken account. Usually, the artificial radionuclides exist at surface of soil in the early stage of accident. In the later stage, the radionuclides would be migrated into the soil, and it can be assumed as the exponential distribution at vertical direction. If there are no climatological or artificial reasons, the assumption would be well fitted. However, the radionuclide distribution can be varied by many reasons such as soil type, density, climate and human activity. Because the depth profile affects the dose rate and evaluation of radioactivity, it is very important to assume the depth profile more exactly. To evaluate the effect by depth profile, it is necessary to make or have the reference site with known activity. However, it is difficult to have the reference site because of homogeneity of radioactivity. Therefore it is necessary to make the reference site with known activity. Considering radioactive waste, it is difficult to make the reference site with artificial radionuclides such as cesium-137. Therefore, we evaluated the depth profile function using naturally occurring radionuclide such as potassium-40. In the case of potassium-40, because it has higher gamma energy than cesium-137, the density of media would be higher than soil. In this research, we assumed the big concrete disk with known potassium activity, and we evaluated the possibility as the reference disk using MCNP simulation comparing the cesium in soil. Using the results, the several concrete disk can be made with variety of activity, and then we can make the different depth profile by arranging the disks. If we can make variable depth profile function with the reference disks, we can evaluate the uncertainty of in-situ gamma spectrometry by depth profile.

Keywords: *In-situ gamma spectrometry, depth profile, radioactivity*

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Poster Session T2

PS2 (T2.5-0963)

Comparison of ^{222}Rn and ^{220}Rn concentrations in earthen dwellings in rural villages in the coastal region of Kenya

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Among the many radon isotopes, ^{222}Rn is the most well-known hence the name radon is often used in its reference. ^{222}Rn occurs in the decay chain of ^{238}U, has a half-life of 3.82 days and is a recognized carcinogen. Another radon isotope, ^{220}Rn (thoron), is gradually gaining prominence especially in Africa and Asia. ^{220}Rn is formed in the decay chain of ^{232}Th; compared to ^{222}Rn, it has a significantly shorter half-life of 55.6 seconds. The main source of the isotopes is the soil. Since dwellings especially in developed countries are for the most part constructed using non-soil materials and have concrete foundations, ^{220}Rn is largely considered of little radiological concern as it is thought to decay within the soil. But most dwellings in rural Kenya for instance are made of soil and as such, ^{220}Rn can potentially escape from soil into indoor air. In some parts of the world for example India, China and even Germany, dwellings constructed primarily using soil-based materials have been observed to have higher ^{220}Rn levels relative to ^{222}Rn. In this paper, we compare ^{222}Rn and ^{220}Rn levels in earthen dwellings in rural villages in the coastal region of Kenya

***Keywords**: Radon, Thoron, Kenya*

ACKNOWLEDGMENTS
Much appreciation to the Institute of Radiation Protection, German Research Center for Environmental Health, Munich, Germany for providing radon-thoron detectors used in this study, and to Dr, Jochen Tschiersch and Dr. Oliver Meisenberg for their support.

Poster Session T2

PS2 (T2.5-0989)

Survey of Indoor Radon Concentrations in Dhaka City, Bangladesh

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Radon, a naturally occurring radioactive gas, decays into progenies that move into human lung through breathing. These radionuclides release α-rays affecting lung tissues and thus cause lung cancer. In Bangladesh, a country having higher lung cancer prevalence[1], it is important to formulate a safety baseline on environmental radioactivity of radon because of elevated indoor radon concentrations[2]. This study performed a preliminary survey on indoor radon (Rn-222, Rn), thoron (Rn-220, Tn) and gamma radiation in capital Dhaka city, Bangladesh. The objective of this study is to determine the concentration level, behavior of indoor radon and thoron and then to estimate annual effective dose for radon and thoron in living environment in Dhaka city. Additionally, this research focused indoor external dose estimation for the city. In this survey, three categorized house types were considered-(a) modern: concrete, tiles and stone built having higher ventilation (b) apartment: concrete built with lower ventilation and (c) traditional: mostly brick built, sometime concrete with extremely low ventilation. Both short-term (continuously 1-hour interval, for 1-6 days) and long-term (integrated for around 1-year) measurements were done in 3 houses and 42 houses respectively in Dhaka city for characterizing indoor radon and thoron concentrations. RAD7 (Durridge Co. Inc.) and RADUET (Radosys Ltd.) were used respectively for these two measurements. Investigated rooms were kept closed during short-term estimation to avoid human activities. Integrated gamma dosimeter (RPL Dosimeter, Chiyoda Technol Corporation) was set with each RADUET for assessing indoor gamma dose. A working level monitor (WLx, Pylon Electronics Inc.) was used for determining radon and thoron progeny concentrations.

As preliminary data in Bangladesh for 42 multistoried houses of Dhaka city, the highest levels of indoor radon and thoron were estimated as 22±9 Bq/m^3 and 39±34 Bq/m^3 respectively in the short-term measurement. Although the average concentrations of indoor radon and thoron were found to be similar levels individually, estimated equilibrium factor (F= 0.5-0.8 for Rn; and F= 0.01-0.03 for Tn) varied with structures and ventilation facilities of investigated houses. The annual mean thoron concentration was derived as 16±11 Bq/m^3, which is around double than radon of 8±5 Bq/m^3. Thus, thoron exhibits around half annual dose (0.07 mSv y^{-1}) of radon (0.18 mSv y^{-1}) in indoor environment following dose estimation method[3]. Traditional house type was found to be contained with elevated thoron levels compared to other house types. Higher thoron concentrations were observed for many houses during long-term survey even though the detector was positioned more than 50 cm from the wall, which exhibits a greater thoron risks; because Bangladeshi people tend to sleep closed to wall. Again, in case of external dose estimation in indoor environment, the range was achieved as 1.1-2.1 mSv y^{-1} with an arithmetic mean of 1.5 mSv y^{-1} which is higher than calculated total internal dose of 0.25 mSv y^{-1} and also ICRP suggested level of 1 mSv y^{-1}. Thus, following the obtained results, this survey formulates several new research scopes in Bangladesh on environmental radioactivity- (I) systematic investigation with reliable detectors is strongly needed, previous studies determined extremely higher levels of radon than this survey; (II) standardization of equilibrium factors for both radon and thoron; is important to calculate presumable internal dose estimation in Bangladesh, as it varies with indoor environment, and (III) building materials could exhibit higher radiation exposures for the residence and in addition, this research could not find out reasons for elevated internal and external dose in many cases; therefore an investigation for factors (exhalation rate, ventilation rate and so on) that affect indoor concentrations and behaviors of radon and thoron is thus essential.

Keywords: Radon & thoron concentration, Dhaka city, Internal and external dose.

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PS2 (T2.5-1050)

Determination of the Radioactivity Concentrations of Natural Radionuclides in PM_{10} Aerosols (2015) and the Associated Inhalation Annual Effective Radiation Dose to the Public at Jeju Gosan Site, Korea

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The Gosan site in Jeju, Korea serves as a natural background site for characterizing the air pollution in Korean peninsula since the atmosphere in the area is relatively very little affected by artificial airborne matters [1]. As an alternative method, we have measured, using the ICP-DRC-MS, radioactivity concentrations of ^{40}K, ^{232}Th and ^{238}U contained in the atmospheric PM_{10} aerosols which were collected at the Jeju Gosan site during the year of 2015. A total of 127 samples have been analyzed, of which 7 samples are those collected during Asian Dust days, 62 samples collected during normal weather days (no-event days), and the remaining samples collected during the days of haze (10 samples) and fog-mist (48 samples). The mean mass concentration of atmospheric PM_{10} aerosols was 41.54 ± 32.75 μg/m^3. During the study period, the mean concentrations of ^{40}K, ^{232}Th and ^{238}U were 27.31 ± 31.32, 59.78 ± 113.52 and 39.71 ± 32.31 pg/m^3, respectively. The radioactive concentrations of ^{40}K, ^{232}Th and ^{238}U were 7.23 ± 8.29, 0.24 ± 0.46 and 0.49 ± 0.40 μBq/m^3, respectively. They are lower than the world averages for the mean atmospheric activity concentrations of ^{232}Th and ^{238}U, which are 0.5 and 1.0 μBq/m^3, respectively [2]. However, the Asian Dust period values were higher than the world average. During Asian Dust days, the mean activity concentrations of ^{40}K, ^{232}Th and ^{238}U were 27.98 ± 14.64, 1.60 ± 1.16 and 1.31 ± 0.68 μBq/m^3, respectively. They are respectively 5.50, 9.41 and 2.85 times higher compared to the normal days. The ^{232}Th/^{238}U activity ratio of totality was 0.49. The ^{232}Th/^{238}U activity ratio of Asian Dust day value was 1.22, which was higher than other atmospheric phenomenon period ratios.

We are analyzing five-day backward trajectories of the air inflow into the Jeju Gosan site using the HYSPLIT4 model of NOAA. The air movement route was classified into Sector I (China continent), II (Korean peninsula), III (Japan) and IV (North Pacific Ocean). The values of sector I were higher than those of other sectors. The mean activity concentrations of ^{40}K, ^{232}Th and ^{238}U of sector I were 11.23 ± 10.78, 0.43 ± 0.68 and 0.65 ± 0.50 μBq/m^3, respectively.

The inhalation annual effective radiation dose (E_i : default mode F) to the public due to natural isotopes of the atmospheric PM_{10} aerosols was in the range 17.43 ~ 83.03 nSv/y, depending on the age group. It is obvious that ^{232}Th is the main contributor to the inhalation annual effective dose. The radioactive concentrations of ^{232}Th was found to be responsible for the total dose. On the other hand, ^{40}K and ^{238}U were found to slightly contribute to the total dose. The inhalation annual effective radiation dose of Asian Dust days PM_{10} aerosols was in the range 113.97 ~ 544.66 nSv/y, which was higher than other atmospheric phenomenon period ratios.

Keywords: PM_{10} aerosols, natural radionuclides, inhalation annual effective radiation dose

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PS2 (T2.5-1052)

Using sediment records to evaluate historical discharges from a nuclear facility in Sweden

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Keywords: cesium, cobalt, europium, plutonium, americium, curium, sediments, dating

Radioactive releases from Swedish nuclear facilities are reported to and supervised by the Swedish Radiation Safety Authority (SSM). One of these nuclear facilities is Studsvik, located in the Trosa archipelago, about 100 km SW of Stockholm. Data on annual liquid radionuclide discharges from this facility are available from 1959 to present, which gives an excellent opportunity to examine behavior of anthropogenic actinides in the marine environment. The recipient bay, Tvären, has high sedimentation rates (~1cm/y) and undisturbed sedimentation bottoms prevail, which enables detailed sediment chronology studies. By comparing inventories of different radionuclides in the sediment cores and the discharge data combined with sediment dating, a validation of the discharge records can be done.

We will present radiometric sediment core data for a suite of radionuclides; γ-emitting nuclides: ^{60}Co, ^{134}Cs, ^{137}Cs, ^{152}Eu, ^{154}Eu; α-emitting nuclides: ^{238,239,240}Pu, ^{241}Am, ^{243,244}Cm and β-emitting nuclides: ^{55}Fe, ^{63}Ni, ^{90}Sr. The sediment cores have been dated using a ^{210}Pb model assuming constant ^{210}Pb flux and initial concentration. The obtained actinide data show high sediment activity concentrations for ^{238}Pu and ^{241}Am (up to 56±2 Bq/kg and 65±7 Bq/kg dry mass, respectively), and unique ^{238}Pu/^{239,240}Pu and ^{241}Am/^{239,240}Pu activity ratios (0.2-1.6 and 0.4-1.5, respectively). ^{234}U/^{238}U and ^{235}U/^{238}U activity ratios suggest natural origin of uranium isotopes in Tvären. Our results indicate that the sediments are suitable for reconstructing the history of the nuclear operation discharges in the marine environment. For instance, the results indicate that less than 5% and about 25% of released amount of ^{137}Cs and ^{60}Co, respectively reside in the bay sediments.

PS2 (T2.5-1113)

Analysis of Long-Term Trend in Environmental Radioactivity using Non-parametric Statistical Technique

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In Korea, the Nuclear Safety Act, Article 104 (Preservation of Environment) stipulates that the installer or operator of facilities shall conduct the survey of radiation environment and the evaluation of the environmental impact for the surroundings of the facilities. The ultimate purpose of this survey is to ensure the health and safety of the general public by ensuring that the radiation dose is kept below the regulated limit. In addition, it would be more useful if we could confirm that meaningful changes are not found in comparison to the historical data (temporally) and comparative points of the facilities (spatially). A common characteristic of environmental radioactivity monitoring data is that it shows asymmetric and mixed distribution. This is the reason why interpretation of these data is difficult.
In this study, the long-term trend analysis of environmental radioactivity was performed by LT-ERAP (Long Term Trend-Environmental Radioactivity Analysis Program). LT-ERAP was developed using the non-parametric statistical technique (Mann-Kendall tests) which is suitable for assessing radiological data, and long-term trend of environmental data. Environmental data is extremely rare to be normal. The non-parametric Mann-Kendal test is suitable for cases which populations are not normally distributed and commonly employed to detect monotonic trends in series of environmental data. LT-ERAP calculates τ coefficient, S statistics, z-values and p-values, and intercepts and slopes of the trend line using Mann-Kendall test method and presents the results in three ways: 'Increasing', 'Decreasing', and 'No trend'.

Sample media are soil, surface seawater, and seabed sediment. Soil was collected from the environment around nuclear facilities and surface seawater and seabed sediment were collected from the area of each facility's discharge. Anthropogenic radionuclides such as ^{90}Sr, ^{137}Cs, and $^{239+240}$Pu were analyzed in each sample. The period for each radioactivity trend analysis is from 1998 to 2018. The analysis excludes a small number of data due to a short period of time. As a result, ^{90}Sr, ^{137}Cs, $^{239+240}$Pu for each sample media showed 'decreasing' or 'no trend' in each site during the overall period. These results showed that anthropogenic radionuclides did not show a long-term trend, 'Increasing' in the environment by operating nuclear facilities. This study is expected to be useful in assessing its impact on the environment. In addition, for an accurate assessment of the effects on the environment, other factors should be considered; such as emissions from the facilities and behavior in the environment including the results of long-term trend analysis.

Keywords: Environmental Radioactivity, Long-term trend, Anthropogenic radionuclides

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PS2 (T2.5-1153)

Radium Deposition and Contamination of Agricultural Lands due to Oil and Gas Wastes in Nigeria

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In this study, the radiological risks associated with oil and gas wastes generated from oil and gas exploration in Nigeria were assessed. Soil samples were collected from agricultural lands near oil and gas production sites. Radium content C_{Ra} in the sample was determined by high-resolution γ-spectroscopy. While Uranium and Thorium are not soluble in water, their radioactive decay product, radium and some its decay products are soluble and they find their way into groundwater and farmlands. Radionuclide concentration in the samples ranged from low to high levels. The average concentration of radium in the agricultural lands is estimated to be 10.57Bq/g. The radium equivalent activity (Ra_{eq}) values for all samples were lower than the maximum permissible limit (32Bq kg^{-1})

Keywords: radium-226, radium-228, radiological risk, radioactive waste, oil and gas waste, oil exploration, agricultural lands,

PS2 (T2.5-1202)

Monte Carlo Simulation of Site-specific Background Model for Radiation Portal Monitor

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In domestic airports and ports, radiation portal monitors (RPMs) have been installed and operated for detecting illegal radioactive materials in a cargo container. In operation of the RPM, however, a false alarm could occasionally occur due to the fluctuation of background radiation; hence, optimization of the site-specific alarm criterion is essential to decrease the false alarm rate and improve detection probability. Although repetitive tests with various source conditions are required, the availability of these tests on the site is very limited due to 24/7 operations of the RPM at airports and ports. Monte Carlo simulation, on the other hand, could be the best option to estimate the performance of the RPM in various conditions for optimizing the site-specific alarm criterion. In the present study, we precisely modeled two commercial RPMs and analyzed the background radiation at the site where the RPMs installed, in order to develop the realistic site-specific Monte Carlo simulation model. Background radiation contributing to the RPM signal mainly comes from radioisotopes of ^{238}U and ^{232}Th series as well as ^{40}K in the soil [1–3]; therefore, their nuclide activity ratios were calculated based on the measured energy spectrum by a high-purity germanium (HPGe) detector and then applied in the background simulation model. The simulated background spectrum from MCNP6 Monte Carlo simulation was compared with the experimental data, which was measured by the polyvinyl toluene (PVT)-based plastic scintillator, installed in the commercial RPM, as shown in Fig. 1. The realistic Monte Carlo-based background model will contribute to developing the site-specific alarm criterion, which depends on the radioactivities of the soil at the specific area and the characteristics of each RPM.

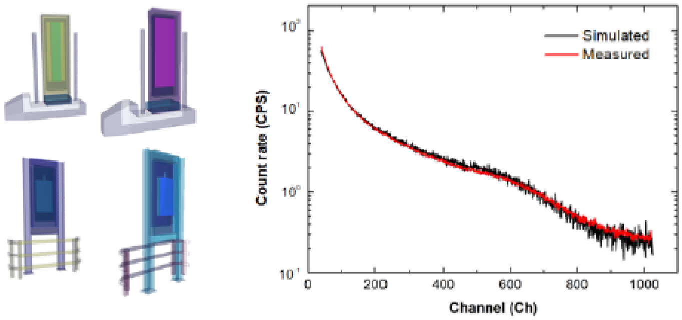

Fig. 1. MCNP models of RPMs (left) and comparison of simulated and measured energy spectrum (right).

***Keywords**: Site-specific Background Model, Radiation Portal Monitor (RPM), Monte Carlo Simulation*

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PS2 (T2.5-1211)

Reporting absorbed dose rate in air from Southern part of Ibaraki prefecture related to Fukushima Daiichi Nuclear Power Plant accident

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Fukushima Daiichi Nuclear power plant (F1-NPP) accident that occurred in March 2011which dramatically changed the distribution of environmental radioactivity in eastern Japan. Due to this accident enormous amount of radio-caesium released into the environment were estimated as 6-20 pBq [1]. After 10 years of F1-NPP accident, radio-caesium (^{134}Cs ($T_{1/2}$ = 2.06 y) + ^{137}Cs ($T_{1/2}$ = 30.17 y)) remains major concern from a radiological safety perspective. Ibaraki prefecture located ~ 180 km southeast of the F1-NPP have been selected to study the dose rate distribution in air. Therefore, the measurement of absorbed dose rate in ambient air were carried out using vehicle mounted 3" x 3" NaI(Tl) scintillation spectrometer (Car-borne survey) in August 2020. The absorbed dose rates in air ranged from 17.72 to 92.63 nGy h^{-1} with an average 41.78 ± 6.57 nGy h^{-1}. The estimated contribution of artificial dose rate in the air were vary from 0.02 to 34.96%. The annual external effective dose ranges from 0.1 to 0.56 mSv. The detailed information of the radiation survey will be discussed during the presentation.

Keywords: *Fukushima Daiichi Nuclear Power Plant accident, Absorbed dose rate in air, Radiocesium, Car-borne survey*

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PS2 (T2.5-1223)

Methods for improving the detection capabilities of environmental radioactivity measurements in the light of increased atmospheric radioactivity levels in 2020

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In 2020 several incidents have occurred that have environmental radiation protection implications. In April elevated radioactivity concentration of ^{137}Cs was measured across Europe as a consequence of the forest fire outbreaks in the Chernobyl exclusion zone. Furthermore, in early June the detection of various artificial radionuclides (such as ^{60}Co, ^{134}Cs, ^{137}Cs and ^{103}Ru) was reported by several Northern European countries. These events showed the continuous and still current need for reliable radioactivity measurements to supply basic environmental radiation monitoring purposes, such as providing information and data for dose assessment purposes and supporting decision making on the necessity of actions for public protection. It must be noted, however, that the increase in atmospheric radioactivity levels as a result of these events were generally very low, which made it necessary to use highly sensitive measurement techniques or to endeavor improvement of detection possibilities of quantitative analysis.

Methods will be presented for the improvement of detection capabilities of environmental radioactivity measurements, with which orders of magnitude improvement in the value of the characteristic limits (decision threshold, detection limit) may be achieved. These approaches were investigated with considering the aspects of practical applicability and the data supply obligations to be met (e.g. the necessity for accurate but also rapid data provision on the levels of radiation and environmental contamination in order to support decision making about the need to take protective actions). The proposals aiming the reduction apply equally to the

- conditions of environmental sampling (e.g. the duration of the sampling period and the volume or mass of the collected samples),
- measurement conditions (e.g. acquisition time and measuring system properties such as detection efficiency),
- evaluation (e.g. in case of gamma-ray spectrometry, the computation of the characteristic limits from the output of spectrum analysis depending on the specific spectroscopic conditions).

Since detection limit is the function of not only the measurand itself but of its associated uncertainty, aspects of reduction of uncertainties in environmental sampling measurements will be also presented.

***Keywords**: Radioactivity measurements, Monitoring strategies, Artificial radionuclides, Detection limit*

PS2 (T2.5-1231)

A transport of inorganic particulate matter in size-classified aerosols attached by 7Be in the atmosphere in Osaka, Japan

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Atmospheric ^{7}Be ($T_{1/2}$ = 53.3 d) is a natural cosmogenic radionuclide that is continuously produced in the upper troposphere and lower stratosphere by spallation. After ^{7}Be is produced, it soon attaches to aerosol and is transported to the lower atmosphere. Many researchers have observed the relationship between ^{7}Be and particulate matter (PM) and note that the concentration and variation of PM can alter the behavior of ^{7}Be and its atmospheric distribution [1, 2]. However there have been few studies that have estimated the size distributions of atmospheric aerosols attached by ^{7}Be. It is well known that nitrate (NO_3^-), ammonium (NH_4^+), sulfate (SO_4^{2-}) and sea salt representing a 1:1 ratio of sodium ion (Na^+) and chloride ion (Cl^-) are significant constituents of the PM [3]. In this study, we investigated the relationship between ^{7}Be concentration and the major chemical composition of PM in size-classified aerosols, in order to investigate the main inorganic species of PM in atmospheric aerosol particles as a carrier of ^{7}Be. Every 2 weeks, aerosols were collected on the rooftop of building No. 16 located in our university campus at Daito, Osaka (34.71 °N, 135.64 °E, the height above ground level, 31 m) in the summer of 2020 by using low-pressure cascade impactor (12 stages with backup filter, Tokyo Dylec Corp., Japan, model LP-20) with a constant flow rate of 20 L/min. The cascade impactor had 12 multi-jet stages, it can classified aerosol into 12 sizes with the cut-off values for the impactor at 50% collection efficiency of 11, 7.8, 5.2, 3.5, 2.1, 1.2, 0.7, 0.49, 0.3, 0.2, 0.12, and 0.06 µm in aerodynamic diameters. The samples for radioactivity measurement were prepared by filtration of $Be(OH)_2$ form and we measured its activity at gamma-ray energy with 477 keV by using a HPGe detector (GX2018, Canberra). For water-soluble inorganic ions were measured by ion chromatography (DIONEX, ICS-1100). Our results during June 6 – 19, 2020 show that NH_4^+ and SO_4^{2-} would be the potential medium for ^{7}Be and these inorganic species can be traced and predicted based on the behavior of ^{7}Be. The accumulated measurement data during summer of 2020 will be analyzed and reported.

Keywords: *Beryllium-7 (^{7}Be), inorganic ion, particulate matter*

ACKNOWLEDGMENTS

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PS2 (T2.5-1242)

Development of lifecycle management methodology for the radiation portal monitor (RPM) in airport and seaport

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In this study, we performed a preliminary study for developing the lifecycle management methodology of radiation portal monitor (RPM) and attached equipment. As a result of documentary survey and analysis of RPM maintenance data, poly vinyl toluene (PVT) and photo multiplier tube (PMT), main parts of RPM, are relatively more stable than other parts. Especially, in case of PVT, no specific lifespan is recommended but for PMT, 5 years are recommended as lifespan to maintain accuracy of detection and improve measurement efficiency. In order to reduce the RPM operating failures, it is recommended that the operation computer which is directly connected main equipment, is anticipative replacing to an industrial computer.

After 2018, malfunctioning rate is remarkably decreased because operation system is improved based on operation and maintenance experience. For stable operation of RPM, it is necessary to perform temperature and humidity monitoring of the place where the RPMs are installed, because these two factors are main cause that can occur malfunctioning. Preparing spare parts of equipment and simplification of parts in the attached equipment are also important for stable operation. Subsections of maintenance report have to be more detailed to get more specific data for follow-up research.

Keywords: *Radiation Portal Monitor, PVT detector, lifecycle*

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PS2 (T2.5-1257)

Estimation of thoron exhalation rate using LR-115 based passive technique

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SSNTD(Solid state nuclear track detector)-based sealed can technique has become outdated because of its inability to discriminate the radon and thoron contributions .A simple way to selectively record the radon or thoron contributions is to make use of their significantly different diffusion lengths in air [1]. When thoron needs to be selectively measured, films are mounted at two different distances. The farther film records only radon and its progenies and the counts can be subtracted from the nearer film. In this work, the thoron exhalation from soil samples is estimated by the above technique of mounting LR-115 films at two different distances. The work deals with optimizing the distance of the LR-115 films from the sample surface so as to record maximum track density for thoron. This depends on the thoron diffusion length and the alpha energy sensitivity of LR-115 film. The track density is then used to estimate the thoron exhalation rate. The present method is based on diffusion mode and not on flow mode (like RAD7) where if the pump flow rate is low, the decay of thoron in the loop has to be accounted for.

The SSNTD film used in the present study is the LR-115 film (composition –Cellulose Nitrate).The soil is taken in a container and the films are fixed at a certain distance from the soil surface. Considering a situation in which an exposure time of 6 hours or less will be sufficient to get significant track density on the film, the contribution to the tracks is mainly from ^{220}Rn and ^{216}Po.Soil sample with high ^{232}Th content (Soil1- 33 KBq Kg^{-1}) was chosen for the study. The average track density obtained for Soil-1 for LR 115 films fixed at the heights 1.5, 2.5, 3.0, 3.5, 4.5 and 5.5cm.from the soil surface and for the LR-115 films at height 14 cm. It is found that the track density is maximum at 3 cm height above the soil surface. The total time integrated activity is calculated from the track density by using the species independent calibration factor for the bare mode exposure [2]. The same analysis is carried out for Soil -2 and Soil-3. Since, the thorium activity was relatively lower, the exposure period was increased to 23 hours. The average steady state thoron concentration and hence the thoron surface exhalation rate (J_s) is calculated. (Table-1). The thoron exhalation rates were found to be underestimated by the RAD7 online monitor when the correction factor for thoron decay loss is not applied.

Table-1: Steady state thoron concentration and exhalation rate

Soil	A^{eq} (Bq m^{-3})	J_s (KBq m^{-2} h^{-1})	A^{eq} (RAD7) (Bq m^{-3})	J_s(RAD7) KBq m^{-2} h^{-1})
Soil-1	38113 ± 2164	86 ± 5	11200 ±611	74 ± 4
Soil-2	8050 ± 444	18 ± 1	2342 ± 86	14 ± 1
Soil-3	8442 ± 470	19 ± 1	2532 ± 83	15 ± 1

Key words: *Thoron, Track detector, Exhalation*

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PS2 (T2.6-0444)

Developing a practical calculation tool for skin dose after contamination

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Introduction

This project was aimed at developing a tool to unify calculation skin dose after contamination(s).
In our university and hospital there is a broad range of radioactive substances being used for research, diagnostics and production of radiopharmaceuticals. These activities are spread out over a large campus. Every location is supervised by a local Radiation Protection Officer (RPO) who is supported by and under surveillance of the central - RPO/RPE) of the radiation protection department (RPD).

Aim

When working with radioactive substances it's inevitable that radioactive contamination accidents take place. Last 10 years the RPD has experienced that RPO's are confronted with commotion during accident situations and therefore find difficulties in calculating the equivalent skin dose and reporting this in a comparable and unified manor. The RPD concluded that RPO's need guidance to make correct dose estimations.

Method

An analysis on contamination events last 10 years shows there is a similar course of the stay of radioactivity after contamination accidents. We conclude calculation of skin dose can be done automatically following this course. There are commercially systems available but these are costly and usually demand highly specific information and knowledge. Therefore I designed a simple tool which guides RPO's filling the minimal data to be able to calculate - worst case - integrated skin dose given a certain activity, on numerous radionuclides mainly used in nuclear medicine and given a period of time (taking the effects of decontamination into account). The tool is checked for calculation accuracy and ease of use by RPO's and RPE's of other institutes and is optimized.

Results

I found a way to support RPO's coping with accidents by providing a simple tool for the calculation and reporting of skin dose due to contamination accidents. The system is very easy to use, is protected for changes, is fail proof and gives unity to dose calculations with frequently used radionuclides used on our campus.
Secondly this tool can be used for risk assessments in case of estimating skin dose due to foreseen accidents in planned exposure situations.
Thirdly this system can be used to create awareness to radiological workers and learn what typical skin doses are following a contamination accident.

In a presentation I will give a schematic overview on the course of radioactive contaminations and I will give an overview on the calculation tool as a good result of radiation protection in practice.

PS2 (T2.6-0672)

Exposure to radon of South African underground miners

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Radon is accepted to be the largest contributor to the dose received by the general public. The group most susceptible to radon is the cohort of underground miners [REF 1-4].

South Africa has the highest number of underground miners in the Western World, around 200 000. The presence of uranium in most mines where gold is present, results in high values of radon in many mines. The concentration of radon depends on many aspects, in particular:

- The radium concentration in the rock since this is the direct parent of radon.
- The ventilation in the mines. The better the ventilation, the lower the radon concentrations are likely to be. Hence older mines where ventilation is often not as good are likely to have higher radon concentrations.
- The radon level is strongly influenced by the blasting operations that take place.

The Beir report [Ref 1] includes a review of the major studies of underground uranium miners Eleven cohort studies were summarised, involving a total of 60 000 miners among whom 2 600 deaths from lung cancer had occurred.

According to the National Nuclear Regulator Act (47/1999 in South Africa): Regulations: Safety standards and regulatory practices, annexure 2, the occupational exposure of any worker must not exceed “an(average) effective dose of 20 mSv per year averaged over five consecutive years”. Miners in South Africa are the occupational workers who receive the largest doses, far larger than workers at the Koeberg Nuclear Power station [REF 5].

This contribution will describe a project to investigate best methods to address the exposure of these miners to radon.

Keywords: *Radon; Underground miners; ventilation.*

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PS2 (T2.6-0781)

Quantitative evaluation of the conservativeness in the committed dose concept

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The protection quantity "committed effective dose" is a risk measure of the stochastic health effect associated with an intake of radionuclides into the human body. Radiations from the incorporated radionuclides in the body irradiate organs and tissues over time periods determined by the physical half-life and the biological retention within the body, and may give rise to dose delivery to organs and tissues over many months or years after the intake. The need to regulate exposures to radionuclides and the accumulation of radiation dose over extended periods of time has led to the definition of "committed dose" quantities. For compliance with dose limits, ICRP recommends that the committed dose be assigned to the year in which intake occurred, although the intakes of radionuclides can lead to continuing radiation exposure over many years. For workers, the committed dose is evaluated over the 50-year period following intake. For plutonium-239, for example, the effective dose in the first year after intake will generally be less than 10% of the total dose.

In this study, we develop an approach to quantitative evaluation of the conservativeness in the concept of committed dose from internal exposures for radiation workers from the viewpoint of radiological risk. Committed dose coefficients (Sv/Bq) of selected radionuclides from ICRP Publications were reproduced to obtain annual effective doses rate (Sv/Bq/y) that would be actually delivered each year over the 50-year period. Then, by assuming that a worker at the age of 18 years would continuously take in the same amount of radionuclides each year until the age of 68 years (i.e., 50 years), death probability rate (1/y) was calculated and compared with those indicated in the ICRP Publication 60 where setting the annual dose limit was developed. As a result, it was found that the inhalation (type S) of H-3, C-14, and Sr-90 show that the committed dose concept overestimates the fatal cancer risk by a factor of about 1.6 at most, comparing the rates at the age of 70 years. This overestimation was about 2.2 for plutonium-239 having much longer effective half-life.

Considering sources of uncertainties which contribute at all stages of the calculation for internal dose exposure, such as biokinetic model, transfer coefficients, etc., the magnitude of overestimation could be minor and may appear justified by its relative simplicity and practicality for the radiological protection purpose. Nonetheless, the authors believe that it is important to quantitatively understand the hidden conservatism in the committed dose concept.

PS2 (T2.6-0883)

Comparison of stress responses of human fibroblast cells by low-dose γ-irradiation and heat shock at the molecular level

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Effects of low-dose radiation (LDR) on living organisms, including human is still debated, in spite of LDR being ubiquitous in our environment. One prominent phenomenon of LDR is hyper-radiosensitivity (HRS) which may increase cancer risk. According to our reports[1, 2], the doses causing HRS by LDR of γ-rays from a ^{60}Co source was 7.5 mGy and alterations in several genes specific to LDR-irradiated cells were suggested by whole genome sequencing (WGS). In this study, we analyzed the change of expression patterns of stress-related proteins like hsp60, hsp70 and hsp110 at the protein and mRNA levels by 42℃ treatment and by irradiation of γ-rays on the cells and compare the stress responses by heat shock and irradiation. In addition, the relation of the stress-related proteins and products of the genes like *FGFR3* (Fibroblast growth factor receptor and *PDGFRA* (Platelet Derived Growth Factor Receptor Alpha) those found to be encoded by the genes altered by LDR-irradiation.

Keywords: *low-dose radiation, human fibroblast cells, stress protein*

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PS2 (T2.6-1003)

Harmonization of exposure estimation procedures in the cohorts participating to the international pooled analysis of uranium processing workers (iPAUW)

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Background: The International Pooled Analysis of Uranium Processing Workers project (iPAUW) will assess the potential long-term health risks of gamma ray, long-lived radionuclides, radon and radon progeny exposure among uranium processing workers from different countries. Joint analyses of data provided by different cohorts need consistent exposure assessment, i.e. the assessed dose for two individuals from two different cohorts should be similar if their exposures are similar. Internal exposures are typically estimated based on information on incorporated radionuclide(s), intake pattern and pathways and physico-chemical forms of contaminating compounds. Some exposure monitoring measurements are individual-based but in other instances, individual exposures are estimated using job-exposure matrices (JEMs). The estimation of exposures to external gamma radiation, internal emitters and radon decay products (RDP) requires knowledge of exposure period(s) and often this is extrapolated from historical data (e.g., extracted from facility records). Moreover, many bioassay measurements or external doses are reported as "below detection/reporting level", which could produce bias in dose estimates notably because of differences in reporting levels between different facilities. Lastly, dose assessment protocols may differ between cohorts on a number of factors (due to inherent subjectivity when interpreting the available data (or lack thereof)). All these aspects lead to systematic deviations in dose estimates between cohorts. Since some parameters (e.g. duration of exposures, treatment of data reported as "below detection/reporting level") are based on subjective judgment, rather than scientific evidence, review and harmonization of the radiation dosimetry between individual cohorts is warranted.

Objective: Determine the most suitable protocol to assess doses from exposures to radon, RDP, long-lived radionuclides, and from external exposures for the cohorts of the iPAUW project, considering the current state of knowledge and available data. This harmonization is critical for conducting pooled risk analyses within the iPAUW.

Results/conclusion: This presentation will focus on the methodology followed by the iPAUW Dosimetry Group during this project to harmonize the exposure estimation procedures. This work will be divided in several steps:

1) Review the available data in different cohorts along with the dosimetric protocols which were used to reconstruct various radiation exposures.
2) Discuss how to build a harmonized protocol for the joint analyses, based on the review results and available data.
3) Define the harmonized procedure to assess doses from external exposures as well as exposures to radon, RDP, and long-lived radionuclides.
4) Calculate a new set of doses for the different cohorts using the harmonized protocol.

***Keywords**: Dosimetry, epidemiological studies, ionizing radiation*

PS2 (T2.B-1072)

How Individual External Dose for the Public is Measured, Accessed and Communicated during Post-Accident Recovery: Experiences and Lessons Learned from Fukushima

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The Fukushima Daiichi Nuclear Power Plant accident occurred on 11 March 2011 released radioactive material into the atmosphere and contaminated land in Fukushima and the neighboring prefectures. During post-accident recovery phase, it is imperative to accurately understand or estimate realistic individual external doses so that individuals can make informed decisions based on their radiological protection to return to or live in the affected areas. Accurate information on individual external doses is also needed by the authorities. Measurements of individual external doses would help the authorities to grasp the dose distribution of the population and to aid in deciding the need for additional large- or maybe small-scale protection measures. Since 2013 the authors used a semiconductor silicon personal dosimeter called "D-shuttle" along with the Global Positioning System (GPS) and Geographic Information System (GIS) to understand realistic individual external doses and to relate individual external doses, ambient doses, and activity-patterns of individuals in the affected areas in Fukushima. To date approximately 300 residents participated in our study. Our measurement data suggested that the individual external doses measured by D-shuttle are generally much lower than those determined using a simple model with airborne-based ambient dose data [1, 2]. The results provide a valuable contribution to understanding realistic individual external doses, and the corresponding time-activity patterns and airborne monitoring air dose rate, which can be useful for estimating future cumulative external doses following the return of residents to their homes in the former evacuation areas. In addition, we have conducted face-to-face interviews to investigate the participants' responses to their measured individual external dose information when we returned and explained the measured data to each participant [3]. Some responses are as follows. Many of respondents considered the radiological conditions important to their life after returning their home. Respondents generally agreed with usefulness of personal dosimetry such as D-shuttle to understand their own external dose characteristics and levels, but the attitudes towards the measurement data were different. The presentation will discuss and emphasize how individual external dose for the public is measured, accessed and communicated during post-accident recovery based mainly upon the authors' experience in measuring, assessing and communicating individual external doses in the affected residents and areas in Fukushima.

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PS3 (T3.1-0103)

Medical monitoring of nuclear industry personnel as a part of the radiation safety system

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Ensuring the professional reliability of nuclear industry personnel is an important element of the radiation safety system. At nuclear facilities, activities that monitor and maintain the health of workers are an effective tool for managing the risks of contingencies due to the human factor.[1]

In the Russian Federation, medical support for nuclear industry personnel, including obligatory periodic medical examinations carried out by the Federal Medico-Biological Agency. An analysis of the results of medical examinations of employees of Rosatom State Corporation operating under the conditions of ionizing exposure from 3.25 to 1.62 mSv per year was carried out.

In 2015-2017 among employees of Rosatom State Corporation , the incidence (the number of cases per 1000 people) with a diagnosis established for the first time in life decreased from 73.9 to 68.1. It should be noted that a similar indicator in the Russian Federation increased from 778.2 to 788.9. The structure of the identified diseases was represented by the following groups: circulatory system diseases (19.1–13.8%), eye diseases (10.9–13.2%), endocrine system diseases (9.2–11.4%), diseases of digestive organs (8.8-6.5%), diseases of the musculoskeletal system (8.9-4.5%), diseases of the genitourinary system (8.8-8.6%). The share of cancer was 4.6 - 4.0% of cases.

Thus, the incidence rate with the diagnosis established for the first time in life among employees of Rosatom State Corporation is lower than that in Russia as a whole. This indicates a higher level of healthcare organization at the FMBA of Russia, which ensures the necessary level of labor safety at nuclear facilities. The obtained data on the structure of the identified diseases will be used to implement specialized preventive programs.

Keywords: medical examinations, radiation safety system, nuclear industry

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PS3 (T3.1-0124)

Development of Guidelines on Radiation Protection for the Lens of the Eye in Japan

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From July 2017 to February 2018, the subcommittee of radiation protection of the lens of the eye of the Radiation Council, Japan discussed implementation of the ICRP new dose limit to the lens of the eye for Japanese regulations related to radiation protection for radiation workers and published the subcommittee report. The Council judged that it is appropriate to revise current Japanese regulations, as recommended in the ICRP Statement; equivalent dose limit of the lens should be 20 mSv in a year, averaged over defined periods of 5 years, with no single year exceeding 50 mSv. The Council also recommended to revise the regulations so that the personal dose equivalent with $H_p(3)$ can be monitored to estimate the equivalent dose to the lens of the eye more accurately, but it is not mandatory to measure $H_p(3)$ if it was possible to estimate the equivalent dose of the lens adequately. In addition, there are various exposure situations such as the medical sector, therefore, it was expected that related academies would prepare guidelines or some documents. The Council recommended to take necessary measures for related regulations with reference to the subcommittee report.

In December 2018, the Ministry of Health, Labor and Welfare (MHLW), Japan, which has regulations related to occupational health started to discuss implementation of the dose limit and the monitoring of the lens of the eye recommended by the Council. The regulations will be revised in early 2020 and enforced in early 2021.

In addition, studies on application of the new dose limit for the lens of the eye continue under the Radiation Safety Research Promotion Fund of Nuclear Regulatory Authority (NRA), Japan and the other funds. In 2019, we started to prepare guidelines on radiation monitoring and radiation protection of the lens of the eye in medical sector in cooperation with related academic societies under the Radiation Safety Research Promotion Fund of NRA. In this presentation, we will report mainly on the dose monitoring methods for the lens of the eye and future challenges in Japan.

Keywords: *Lens of the eye, Radiation protection and monitoring, Guidelines*

ACKNOWLEDGMENTS

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PS3 (T3.1-0278)

Development of Software to Compute Structural Shielding Requirements for the Installation of X-ray and Fluoroscopy equipment in a radiography and fluoroscopy room

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Computation of structural shielding requirements for the installation of general-purpose radiographic facilities involves series of complex mathematical calculations which is very demanding in a way. An automated software program was developed to compute the structural shielding requirements for the installation of X-ray and fluoroscopy equipment in a radiography and fluoroscopy (R&F) room. The software program, Automatic R&F Shielding Calculator provides fast and precise calculation of required shielding for medical X-ray and fluoroscopy imaging room. Using values entered by the user and various conditional if, nested if, and else if statements, for and next statements, fitting tables and elements, the software was able to calculate the following: primary air kerma, secondary air kerma, primary barrier thickness, secondary barrier thickness, primary barrier transmission factor, secondary barrier transmission factor and additional barrier thickness for shielding considerations of X-rays generated in the imaging room. In addition to R&F room shielding calculation, the software could compute the requirements of shielding for Rad Room (chest bucky), Rad Room (floor and other barriers) and chest room. Analysis of program calculation and manual hand calculation (based on National Council on Radiation Protection and Measurements recommendation) indicated that the maximum deviation from the hand calculation was 0.02 mm. In many cases, the deference between hand and program calculation was found to be a fraction of a millimeter. This maximum error could be attributed to rounding values in calculation. This was because the software does not round values until final stage of calculations. The minimum deviation obtained was 0.005 mm.

Keywords: *Shielding, air kerma, primary barrier thickness*

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PS3 (T3.1-0278)

Development of Software to Compute Structural Shielding Requirements for the Installation of X-ray and Fluoroscopy equipment in a radiography and fluoroscopy room

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PS3 (T3.1-0517)

Effectiveness of Clinical Imaging Guidelines in the reduction of inappropriate requisitions

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Background; Ionizing radiation has revolutionized the practice of medicine. Availability of diagnostic imaging has prompted inappropriate use over the past century. Clinical Imaging Guidelines (CIGs) have been demonstrated to reduce such inappropriate use. There is insufficient literature on appropriateness of imaging requisitions and effectiveness of CIGs in Uganda. **Objectives;** To determine the level of appropriateness of head CT imaging requisitions pre and post intervention. To determine the effectiveness of CIGs in reduction of inappropriate imaging requisitions. **Methods;** A clinical audit of head CT requisitions with an intervention was conducted at Mengo Hospital. Baseline review of 262 requisitions for October, November, December 2018 was done to determine the level of appropriateness. Later a training on the use of CIGs(I-Guide) was provided to referrers in the months of January, February, March, April. A post intervention review of 154 requisitions was conducted for June, July, and August 2019. The level of appropriateness was measured as a proportion of requisitions with the best scores on the I-Guide and a difference in the two proportions was tested. A percentage decrease in inappropriate requisitions was also determined. **Results;** We found that 53% of 262 requisitions were inappropriate at baseline and 47% of 154 were inappropriate after intervention. We tested for difference of the two levels (proportions) and found them not to be similar. We also found an 11.32% decrease in the level of inappropriateness after the intervention. **Conclusion;** CIGs are effective in reducing inappropriate requisitions of head CT exams and must be provided to referrers to be used during routine care.

Key words; Radiation protection, clinical imaging guidelines, CT

PS3 (T3.1-0617)

Comparison Analysis and Application of Radioactive Material Security Management

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Radioactive sources provide a huge benefit to humanity in many areas such as agriculture, industry, medicine and research. But if they are not controlled properly and are used maliciously, they can cause serious consequences such as permanent injuries. Therefore, the security of radioactive sources has been emphasized worldwide, and many countries have established and implemented security management systems to prevent the malicious use of radioactive sources.

The regulation of Korea's radioisotope security management only applies to categories 1 and 2 radioactive sources. But Code of Conduct on the Safety and Security of Radioactive Sources (hereafter referred to as the 'Code of Conduct') and the IAEA Security Series No.11 Sources (hereafter referred to as the 'Series No. 11') focus on the sources within category 3 (Table 1). While the regulation of Korea focus on unsealed radioactive sources, Code of Conduct and Series No.11 do not. Therefore this paper will study differences between the international frameworks security of radioactive sources and Korean security regulations and make up for the weak points because malicious acts using radioactive sources may occur not only to low activity sources but to unsealed sources. In addition, the adequacy of the Advanced Radiation Technology Institute (ARTI)'s security management system is verified through comparing it to the International Atomic Energy Agency (IAEA) security series.

Table 1. Activities corresponding to thresholds of categories example

Radionuclide	Category1 1000×D		Category2 10×D		Category3 D	
	(TBq)	(Ci)	(TBq)	(Ci)	(TBq)	(Ci)
Am-241	6.E+01	2.E+03	6.E-01	2.E+01	6.E-02	2.E+00
Cf-252	2.E+01	5.E+02	2.E-01	5.E+00	2.E-02	5.E-01
Co-60	3.E+01	8.E+02	3.E-01	8.E+00	3.E-02	8.E-01

Keywords: *Radioactive sources, Security system, ARTI, Code of Conduct, IAEA Security Series No.11*

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PS3 (T3.1-0627)

Calibration and Measurement of Radon in drinking-water for rapid monitoring

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More and more attention were put into the health effect of radon in drinking water in China. it is very time comsuing and tedious handling procedures for traditional measurement of radon in drinking water. for rapid measure the concentration of radon in drinking water, a high speed fluence and recycle system must been accomplished. but a flexiable calibration setup for such system is very important to achieve a accurate results in several minutes. here, a calibration system with different concentration of radon in drinking water was introduced by manual setup. the $BaCo_3$ (Ra) powder was grinded, and then dissolved in the acid with exact concentration. This solution is used as a calibration source for the measurement of radon in drinking water. An measure system component of AlphaGuard, sealed bottle and tap-water bottle were setup. and with the calibration, the concentration of radon in tap-water in Tsinghua campus were monitoring contiously for a long time. The average concentration of radon in tap-water for drinking is 2.57 Bq/L in this study.

keywords: Radon in water; calibration; drinking water

PS3 (T3.1-0661)

The Improvement of Radiation Protection in HANUL NPP 3&4 after WANO Peer Review

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KHNP(Korea Hydro & Nuclear Power) operates 25 NPP(Nuclear Power Plant). WANO(World Association of Nuclear Operators) inspects all Korea NPP every 4 years. WANO recommends NPP several AFI(Area for Improvement). Hanul NPP 3&4 were inspected by WANO PR(Peer Review) for 2 weeks(14MAR2019 ~ 28MAR2019). Here are improvements of RP(Radiation Protection) in Hanul NPP 3&4 after WANO PR.

1. HRA(High Radiation Area) set and release criteria
Before : If dose rate is higher than 1mSv/hr, then set HRA by padlock. If dose rate is lower than 1mS/hr, then release padlock.
Improvement : If dose rate is higher than 1mSv/hr or average 0.8mSv/hr for 3 months, then set HRA by padlock. If dose rate is lower than average 0.5mSv/hr for 3 months, then release padlock.

2. FSAR(Final Safety Analysis Report) Zone 6(designed dose rate is higher than 1mSv/hr) access control
Before : FSAR Zone 6 access is controlled by NPP Operation Team key.
Improvement : FSAR Zone 6 access is controlled by installing padlock device additionally from Radiation Safety Team(Total 109 areas).

3. Radiation information sign(dose rate, air contamination, surface contamination, etc) add
Before : 45 areas show Radiation information sign and update regularly.
Improvement : 200 areas show Radiation information sign by adding more plates and update regularly.

4. Tighten internal exposure examination by WBC(Whole Body Counter)
Before : If nose or mouth is radiologically contaminated, then examined by WBC.
Improvement : If contaminated radiologically above the neck, then examined by WBC.

5. Strengthen NPMOS(Nuclear Power Monitoring & Observation System) in RP area
Before : Radiation Safety Team performs NPMOS 4 times a month.
Improvement : Radiation Safety Team performs NPMOS more than 12 times a month.

6. ALARA(As Low As Reasonably Achievable) committee contents plus
Before : Discuss NPP outage goal dose rate, radiation safety control strategy, team cooperation, etc.
Improvement : Plus, analysis of partner company dose rate result recent 3 years, high risk cases, etc.

Hanul NPP 3&4 discussed radiation protection with WANO PR team and performed a lot of recommendations. By strict application of radiation safety, Hanul NPP 3&4 can manage RCA(Radiation Controlled Area) ahead of international standards. Therefore, Hanul NPP 3&4 could reduce potential radiation safety events.

Keywords: *Nuclear Power Plant, Radiation Protection, WANO PR*

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PS3 (T3.1-0836)

Use of the Graded Approach to Management and Disposal of TENORM in the USA

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With few exceptions, management and disposal of NORM residuals and wastes are not regulated at the federal level in the United States. Rather, a patchwork of state and local regulations, regulatory exemptions, policy and guidance documents, along with a large body of legal determinations are used. While this application of the "graded approach" may not be what planners and regulators have in mind when invoking that term, for the most part things end up being managed within a "safety envelope" for workers and the public that permeates all decisions. However, there have been exceptions where the safety envelope was not adequately defined and where environmental damage has been noted. The few federal regulations that do apply, along with other relevant and appropriate regulations that have a nexus with NORM will be mentioned. Current state regulations, and their varying approaches to management and disposal will be listed. National consensus standards, which are usually followed in lieu of regulations will be discussed.

PS3 (T3.1-0859)

SRReW – A Tool to Help Implement the Graded Approach to Radiation Regulation

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The underlying philosophy of the graded approach to radiation protection is that the resources directed at a particular activity to improve safety should be commensurate with the level of risk the activity presents. This is particularly important for complex users of radiation (regulators, universities, etc) who generally have limited radiation safety resources to cover multiple radiation uses.

While the graded approach is supported in theory, it is rarely applied consistently in practice. Potential reasons for this include differing regulatory frameworks across jurisdictions, very few people with a broad understanding of all radiation uses, a desire for conservatism, and a perception that implementing a graded approach is a complex task.

To assist with implementing a graded approach to radiation practices, the authors have developed the SRReW (Standardized Radiation Resource Weighting) index. When SRReW is applied to a practice the algorithm generates a dimensionless value, enabling users to compare the risk associated with that practice against risks in other practices. Application of the SRReW index will allow resources to be allocated appropriately across a wide range of radiation uses. It is simple to use and does not require an in-depth understanding of all radiation practices in order to categorize each practice relative to each other.

Risk rankings are used in conventional safety to ensure that the correct level of attention and control is applied to each hazard associated with a situation or practice. The development of the SRReW index is intended to differentiate the required regulatory scrutiny to which different activities should be subjected. However, SRReW has wider application, particularly for radiation protection practitioners who are required to allocate limited resources across complex institutions such as universities and radioactive ore processing plants.

Examples of how values of SRReW index are generated will be demonstrated.

***Keywords**: Graded Approach, Regulation, Optimisation*

PS3 (T3.1-0951)

Funeral of deceased patient after administration of radionuclides for therapy purposes; radiation protection recommendation and communication

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Treatment of patients with radionuclides is increasing and new radiopharmaceuticals for these therapies are being introduced in hospitals. Therefore, current national guidelines need to be evaluated, revised and updated.

A goal of this evaluation is to gain more insight in the radiological risk due to patients dying shortly after radionuclide therapy. Information and knowledge about the possible radiological risks is necessary for both hospitals and the funeral industry.

To assess the impact of the funeral after radionuclide therapy the Dutch Authority for Nuclear Safety and Radiation Protection (ANVS) requested the National Institute for Public Health and the Environment (RIVM) to research and estimate the radiological risks associated with the death of patients treated with radionuclides.

With these results the ANVS developed recommendations to update the current national guidelines for radionuclide therapy. Due to the possibility of radiation exposure to employees of funeral homes, the update to the guidelines was collaborated with the Ministry of Social Affairs and Employment (SZW).

Besides the recommendation for the evaluation of the guidelines, the communication within this process is very important. Stakeholder involvement is necessary. The hospitals are used to working with radionuclides, whereas the funeral industry is not. They do have their own industry guidelines when handling radioactive clients but are far from familiar with radiation protection principles.

This presentation will give an overview of the regulatory outcome of the research, the recommendations, the complexity of the process and the communication involved.

PS3 (T3.1-0997)

Assessment of the professional radiological risk at the institute in Tunis

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Background: The mapping of the occupational radiological risk is based on a step of identification of the sources of exposure to ionizing radiation, then on an evaluation of the exposure and finally a hierarchy of the risks whose objective was to propose a plan of organization of the Radiation protection of workers within the establishment. The Salah Azaiez anti cancer center brings together several medical specialties using different types of sources of exposure to ionizing radiation. The workers involved in this exhibition must be protected in accordance with the requirements of the Tunisian Radiation Protection Law and labor code. The purpose of our work was to assess the radiological risk, to identify risky situations and to study dosimetric monitoring of exposed workers in order to propose a radiation protection action plan for workers with solutions adapted to any deficiencies.

Methods: We carried out a multicentric, transversal and descriptive study, based on a pre-established form in accordance with International recommendations and a dosimetric follow-up of personnel working on ionizing radiation at ISA during the year 2018.

Results : We identified several sources of exposure in both nuclear medicine and radiotherapy departments using sealed sources and unsealed sources.There were also radiation emitting devices such as linear accelerators and hybrid gamma cameras. The organization of the radiation protection in these services was ensured by the head of department and a competent person in radioprotection. The design of the premises was in accordance with the CNRP guidelines with shielding and sealing in the standards. The non-compliance was noted in the hot laboratory and the injection room of the radioactive patients. Zoning and solid and liquid waste management have been weak, especially for the nuclear medicine department. No procedures for radiological emergency management were found except for some international protocols, a record of radiological incidents or accidents, and not even an emergency plan. With regard to worker exposure, the average annual dose was 1.75 mSv for workers in the Nuclear Medicine department and 1.39 mSv for radiotherapy workers. Medical monitoring of workers in occupational health was not ensured with the absence of a professional exposure sheet for each worker.

Conclusion : The increasing complexity of techniques using ionizing radiation in medical settings requires the institution to ensure continuous improvement of the quality and safety of practices. Thus the establishment of a culture of radiological safety is fundamental with respect for the principles of radioprotection of workers.

Key-words : X-radiation, Risk, Health, Radioprotection, Radioactivity.

PS3 (T3.1-1048)

Current Status of Recommendations for International Radiation Protection in Republic of Korea

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The Nuclear Safety Act, a law on radiation protection of Republic of Korea, was amended between 1998 and 2002 based on the ICRP60(1990). However, ICRP103(2007) and IAEA GSR part.3(2014) have not yet been reflected. This is because the concept of radiation protection has changed from exclusion, exemption and regulatory intervention to exposure scenario categorization(planning/existing/emergency) and quantified optimization (dose constraints and reference levels) have changed, further discussion is needed about how to implement them.

If the effect of the amendment is not sufficiently considered, it will cause confusion among users. Recently, the Act on the succession of nuclear-related projects and the assigning a delegate of radiation safety officer was amended along with the laws in other areas. The revised decree is written in a simple sentence, but many questions and answers were required to apply it in various user environments.

The Republic of Korea conducted a comparative analysis of the Nuclear Safety Act and IAEA General Safety Requirements by topics. The Nuclear Safety Act was lacked or differed on the categorization, justification, optimization, and exemption levels depending on the circumstances of the exposure. The revision of the Nuclear Safety Act will be implemented sequentially in order to harmonize with the international community. Numbers listed in the Notice of Nuclear Safety and Security Commission "Standards on Radiation Protection, etc." are plan to be updated next year.

Table 1. Reference of Derived limit

Terms	Reference
Calculation method of annual limit of intake and derived air concentration	ICRP 61(1991)
Chemical forms by radioactive isotope, Dose conversion factor	IAEA Safety Series 115(1996)
Reference man	ICRP23(1975)

Keywords: *Regulation, The Nuclear Safety Act, Derived limit, IAEA GSR part.3*

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PS3 (T3.1-1097)

A Review on Experience Alpha Control during Decommissioning of NPP

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One aspect of radiation protection during decommissioning that can be a major consideration is the level of alpha contamination that is present on plant equipment. High levels of alpha can increase the need for airborne radiological controls, and subsequently increase the difficulty of the decommissioning work and potentially have an effect on worker exposures. For some NPP decommissioning projects, the challenge has been to handle a wide range of alpha hazards up to very high levels, sometimes with little accompanying beta/gamma radioactivity. This challenge is primarily due to the fact that very small amounts of alpha contamination or airborne activity can give rise to significant internal dose and it is more difficult to directly measure airborne alpha radioactivity.

The purpose of this study is to review the experiences with Radiation Protection (RP) programs during decommissioning in terms of the types of alpha controls that are used to avoid incidents and to find implications for the decommissioning of Kori-1 Unit in Korea. The Connecticut Yankee and Humboldt Bay nuclear power plants are examples that needed to institute additional radiation protection controls due to an elevated alpha hazard. The Connecticut Yankee (CY) NPP experienced significant fuel failures during several operating cycles. CY made extensive use of Powered Air Purifying Respirators during high alpha risk work. Monitoring for airborne alpha radioactivity was also done in areas near to high alpha areas. The CY RP program, which was used during operations, was upgraded early in the decommissioning. These efforts helped CY to minimize the incidence of internal radiation exposures during the decommissioning. The Humboldt Bay (HB) NPP had even greater alpha-related challenges than CY during their decommissioning as significant fuel failures during plant operations and 30-year safe storage period before dismantlement was begun. Such situation at HB resulted in a higher level of RP controls than had been used at CY. The additional controls are as follows: During contaminated system removal, at least two barriers which could be engineering controls, respirators, glove bags between the contamination and personnel were required. And only mechanical cutting was used to segment systems. Table 1 summarizes the considerations for alpha controls based on experience from CY and HB. The CY and HB plant's experiences illustrate that the RP program needs to consider a number of factors in setting controls that protect personnel should that plant have an alpha risk. And these experiences will be applicable to protect workers even during the decommissioning of Kori-1 Unit.

Table 1. RP Work Controls at CY and HB [1]

Connecticut Yankee	Humboldt Bay
· General area air sampling · Lapel air sampling was used as "Intake Dosimeters" · Whole Body Count (WBC) initiation events identified: - Elevated air or lapel sample result - Contamination identified in personnel air intake areas · In-vitro sampling initiated by elevated WBC - Urine and fecal sampling · Random bioassay selection as a check of the program's effectiveness · Internal dose assessment process proceduralized	· Area decontamination prior to work · Engineering controls such as the following were used: - Area ventilation, Keep surfaces wet - Grout/foam the insides of components and piping prior to cutting and seal pipe ends before transporting to the shipping package · Protective clothing · Respirators · Alpha Continuous Air Monitors · Extensive use of Lapel Air Samplers

Keywords: *Radiation Protection, Alpha contamination, Decommissioning*

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PS3 (T3.1-1175)

Radiation Protection Practice In Ghana

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The practice of Radiation Protection in Ghana started in the **1960's** at the then Physics Department of the University College (now University of Ghana), Legon, on behalf of the Ministry of Defense. In 1961, the Government of Ghana decided to undertake the Ghana Nuclear Reactor Project (GNRP) in order to introduce science and technology into the country to aid in national development to be used for solely research, training and production of isotopes. Effective radiation protection infrastructure development in Ghana started with the promulgation of the Radiation Protection Instrument, LI 1559 in 1993 which established the Radiation Protection Board (RPB) under the Atomic Energy Commission by PNDC Law 308 in 1993. Initially the RPB functioned as both regulatory authority and service provider which amounted to conflict of interest. In the year 2000, the Radiation Protection Institute (RPI) was established to provide technical services to the RPB in line with implementation of the LI 1559. In the year 2015 a new Act, the Nuclear Regulatory Authority Act 895, 2015 was promulgated. The Act established the Nuclear Regulatory Authority (NRA), Ghana as the statutory nuclear regulatory body in Ghana, thus separating the regulatory functions from the operational functions of the Ghana Atomic Energy Commission. The Act also allowed Ghana's to include nuclear power to its energy mix. With the establishment of the NRA, the RPI of the GAEC to undertake the position of a Technical Support Organization (TSO) for the successful implementation of the Act 895 of 2015. The Scope of Radiation Protection covers ionising and non-ionising radiation. Ionising Radiation involves: Personal monitoring of individual workers exposure to radiation and Workplace Radiation Monitoring, Radiation instrument calibration and performance testing, analysis of food and environmental materials, including; air, soil, water, etc, Safe and secure management of radioactive Waste and Radiation Protection and Safety Training programs for Radiation Protection Officers and Qualified Operators. The non-ionising radiation also covers: the principal focus is RF radiation from telecommunication cell sites and magnetic resonance application for medical diagnosis. An average of about 1300 cell sites are monitored annually since 2010. Prior evaluation and assessment of telecommunication cell sites are under taken for an average of 500 sites a year since 2012. There were monitoring of extremely low frequency fields around high-tension electric line within the towns and cities. The School of Nuclear and Allied Sciences (SNAS) was established in 2006 by Ghana Atomic Energy Commission in collaboration with the University of Ghana and the International Atomic Energy Agency (IAEA) to develop and maintain human resource in nuclear and allied sciences. The Department of Nuclear Safety and Security of the SNAS is responsible for the development of human resource capacity in Radiation Protection in the form of Post Graduate Education Certificate (PGEC), M. Phil and PhD programmes in Health Physics and Radiation Protection.

***Keywords**: Radiation, Training, Safety Assessment*

ACKNOWLEDGMENTS
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PS3 (T3.2-0094)

When has it been dangerous to work in the nuclear industry system, now or 60–70 years back? Philosophical paradox

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Starting from the beginnings of the nuclear industry, radiation safety standards (RSS) are being tightened: the dose limit for workers has decreased by two orders, from 1,560 mSv/year in 1925 to 10–20 mSv/year in the 1990s [12]. Logically, mortality from pathologies related to radiation should also decrease, but the situation is not so simple.

In terms of the standard mortality index (SMR), calculated in comparison with the general population nuclear workers in all periods had lower overall mortality ([3–6] and others). This is associated with "healthy worker effect." [7] It would seem that with a decrease in radiation doses, SMR should also decrease, but this is not so. For example, for the united nuclear industry cohort of Great Britain (UKAEA) SMRs in the 1940s—until the mid 1950s. was 0.4–0.6, but then changed to 0.7–0.8 and higher, remaining at this level [2]. A similar trend has been observed for the Mayak Production Association since the mid-1970s. until 2010 [4]. That is, in spite of the improvement of working conditions, technological progress and the tightening of RSS, from a formal position it is becoming less and less safe to work in the nuclear industry system compared to ordinary employment, since the relative mortality from all causes is higher from decade to decade (or a plateau is reached). This is due to a number of possible biases and confounders, for example, the improvement of public health (which reduces the "healthy worker effect"). Nevertheless, such a pattern exists, and for ordinary as well as ordinary-scientific consciousness, it may seem to have causal meaning.

For workers in the nuclear industry, the SMR not only for general mortality, but also for most specific diseases is smaller than for the general population [3–6]. The only systematic reverse exceptions are circulatory pathologies, cancer and leukemia, although mainly for the contingent of previous decades [1]. But for most malignant neoplasms, as well as for general mortality, we were not able to identify a tendency for a decrease in SMR from decade to decade.

Thus, an attempt to link the tightening of radiation safety standards and health effects for nuclear industry workers with counterfactual dependency was not very successful. With one exception, the over time decrease in SMR for lung cancer in a study [9] for the Nuclear Center in Oakridge from 1947 to 1974. It is likely that only for this pathology can systematic counterfactual dependency be observed.

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PS3 (T3.2-0363)

Determining the Local Diagnostic Reference Levels for Radionuclide Bone Scintigraphy at Department of Nuclear Medicine, University College Hospital, Ibadan

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INTRODUCTION: Sensible use of imaging modality requires strict adherence to the tenets of radiation protection; justification, optimization and minimization to ensure that the risk to patients does not outweigh the benefit from the technique. At the core of optimization is the establishment of diagnostic reference levels (DRLs) which are based on relevant local, regional or national data. There are no known data for DRL in Nuclear Medicine in Nigeria.

OBJECTIVE: To determine the local Diagnostic Reference Level (DRL) for Radionuclide Bone Scintigraphy to rule out breach in optimization of radiation protection.

METHODOLOGY: This study was a prospective, cross-sectional study carried out in the Nuclear Medicine Department of University College Hospital (UCH), Ibadan. One hundred and nine (109) patients consented to be part of the study. The data, in this study, were collected from June 2017 to March 2019 and analyzed to obtain the mean and standard deviation of the anthropometric variables, technical parameters, and radiation dose received. Seventy-fifth percentile or (3rd quartile) value of the total mean of the examinations and/or procedures were obtained at 95% confidence interval.

RESULT: The result showed that the mean administered activity, absorbed dose and DRL (3rd quartile value) were 22.5 ± 2.71 mCi, 52.5 ± 6.31 mGy and 24.2 mCi (895 MBq) respectively. Furthermore, the calculated local DRL was found to be larger than values reported in previous studies done in Sudan, United Kingdom, Australia and ICRP standards. However, the lower boundary value of administered activity was found to be 17.1 mCi (633 MBq) which proves to be optimum for image of diagnostic value.

Table 1. Comparison between local DRL (MBq) and international standard value(s)

Radiopharmaceutical	Calculated Local DRLs	Sudan DRLs	United Kingdom	Australia	ICRP range
^{99m}Tc-MDP Bone scan	895	777	600	890	500-1110

CONCLUSION: These guidelines are needed to prevent the travail, needless radiation dose, and unproductive use of time and resources for patients, imaging experts and care service institutions.

Keywords: Administered Activity, Diagnostic Reference Levels (DRLs), Nuclear Medicine, Radionuclide Bone Scintigraphy.

PS3 (T3.3-0025)

Determination Of Target Exposure Index In Digital Chest Radiography Thru Optimizing Exposure Technique Factors

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Introduction Digital radiography (DR) offers numerous advantages such as the wide exposure latitude and image post processing capabilities which provide consistent image appearance even with underexposed and overexposed radiographs. However, this overexposure can go unrecognized with the corresponding needless extra radiation dose to the patient. Moreover, the noise intolerance by the radiologists can lead to exposure creep where technologists tend to increase exposure factors to make sure that images acquired are acceptable to the radiologist.

This study aims to establish the target exposure index (EI_T) for adult digital chest radiography performed in the Las Piñas General Hospital & Satellite Trauma Center thru optimization of exposure technique in order to eliminate dose creep. EI_T is the expected value of EI when an image is optimally exposed, thus, the image is of diagnostic quality at minimum dose possible. It provides feedback to the technologist whether a proper radiographic technique was used

Methodology: The study included medium size (with thickness ranging from 21 cm – 30 cm) patients who had chest radiography in the hospital. Exposure factors, displayed EI and AEC density setting were recorded during image acquisition. Three (3) radiologists evaluated images based on the image quality criteria and method agreed upon by them. The degree of agreement of the three radiologists in their rating was evaluated using the Two-Factor Analysis of Variance (ANOVA). The variances were tested by means of the F test. The average of the EI values of the images with final category of Pass was obtained. This average was recommended to be the target EI.

Results: There were 100 images evaluated. In the process of establishing the EI_T, the percentage of repeat exposures for adult chest PA images vastly reduced from 11% to 1%. Also, an average of 67% and a maximum of 78% patient dose reduction was attained. The degree of agreement among the radiologists in the evaluation of the images showed a value of F of 0.1 for (2,198) degrees of freedom, which is insignificant with respect to the 5 percent α (which is 3.04). This indicates that the raters did not differ significantly among themselves in their ratings of the images. Finally, the value of the recommended EI_T was 11.

Conclusion: This study showed that: (1) optimization of exposure factors is necessary in determining the EI_T; (2) the use of AEC in optimizing exposure factor and consistent production of good quality image is invaluable; (3) establishment of standard exposure technique is important in DR; (4) the use of caliper is important even in DR; (5) it is critical that radiologists, radiologic technologists, and physicists work as a team to assure images are of diagnostic quality at a minimum dose.

Keywords: digital radiography, exposure index, optimization

TABLES AND FIGURES

Table 1

Exposure Technique Factors Used in the Pre and Post Adjustment of the AEC

	kV			mAs		
Period	Min	Max	Mean	Min	Max	Mean
Pre adjustment of AEC*	70	120	91	0.32	7.36	1.57
Post adjustment of AEC	100	100	100	0.32	2.56	0.41

*includes factors using both manual and AEC modes

Table 2

EI Values and Patient Doses in Pre and Post Adjustment of the AEC

	EI			ESAK* (mGy)		
Period	Min	Max	Mean	Min	Max	Mean
Pre adjustment of AEC	5	7302	48	0.01	0.194	0.033
Post adjustment of AEC	7	31	13	0.008	0.042	0.011

*Entrance Surface Air Kerma

Table 3

Statistical Analysis of the Scores

Source of Variation	df	SS	MS	F	α
Rows (Images)	99	32	0.33		
Columns (Radiologists)	2	0.05	0.02	0.10	0.05
Error	198	48	0.25		

Table 4 – Summary of Scores

AEC Density Setting	No. of Images with a Score of				
	1	2	3	4	5
0	0	14	42	3	1
-1	0	9	48	1	2
-2	0	5	51	3	1
-3	0	10	46	2	2
-4	0	16	44	0	0
TOTAL	0	54	231	9	6

Table 5

Range of EI Values Per AEC Density Setting

AEC Density Setting	EI Value		
	Min	Max	Mean
0	7	31	15
-1	10	18	13
-2	9	16	11
-3	9	27	12
-4	7	23	10

Figure 1
No. of Pass Images Per Density Setting

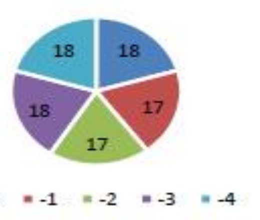


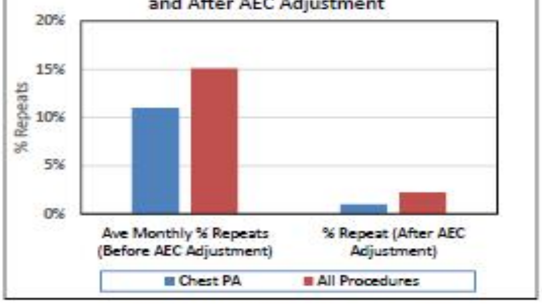
Figure 2

Comparison of Digital Chest X-Ray Images

A- First Digital Chest X-ray Image Using Manual Mode

B – Digital Chest X-ray Image Acquired at -5 AEC Density Setting

Fidgure 3 - Percentage of Repeats Before and After AEC Adjustment



PS3 (T3.3-0469)

About X Ray Equipment Used In Veterinary Practices

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Context

In small animal practices, most of the animals are held by hands: in the best conditions, head is about 50 cm of the X ray tube; for small animals even, when professionals hold it alone, the head is about 10 cm away of X ray tube.

In equine or zoo radiography practices, the mobiles are used by veterinarians most of time kept by hand near the belly or even on the shoulder near the eyes. The eyes are so between 10 to 50 cm maximum of the generator.

Legislation

The X ray generators used in veterinary medicine have not more to be certified with CE medical 93/42 since this directive is only applicable to medical practices ; the old directive 84/539 applied for the veterinary and medical application but has been repealed in 2008.
The answer of the DG Industry is that, when veterinary X ray equipment are not also used in human field, so when they are specific equipment, they should respect 3 others directives:

PS3 (T3.3-0754)

The JRC-Ispra system for radiological protection: an integrated ALARA approach applied in a real case of radioactive waste treatment

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The Joint Research Centre of the European Commission in Ispra (Italy) is a complex nuclear entity. Due to the past and current research experiences, several nuclear and radiological installations still exist on the site, many of which are permanently in shut down or in decommissioning phase. Among others, as an example of installations, there are a nuclear research reactor, a hot cell facility, some laboratories and a radioactive waste management facility. The complexity of the activities carried out requires the implementation of a robust radiological protection (RP) system, which allows from one hand to implement an effective ALARA approach - chiefly in the low dose regime - and from the other hand to facilitate the identification of any deviation of the radiation protection quantities.
Radiation protection constraints have been defined starting from the legal dose limits, and reference levels have been set up accordingly. For all the relevant radiation protection quantities a complete set of reference levels (i.e. registration, investigation and intervention levels) constitute the base of the integrated radiological protection system in force on site.
The reference levels are effectively applied to workplaces and to workers monitoring, including internal contamination assessment (via WBC and RTX analyses), to environmental monitoring including in particular feed and foods (e.g. meat, milk, leaf vegetables) and environmental matrixes (i.e. air particulate, wet/dry precipitations, surface water).
The defined reference levels are applied in daily routine and special activities, providing the basis to set up the alarm thresholds for radiation monitoring devices (i.e. portable and fix instrumentations, and personal dosimeters) and for the authorization of the radiological activities.

In some cases, a flexible approach is required without compromising the coherence of the RP system. A practical example of this approach, is the definition of specific reference levels for the activities of sorting and re-packaging of historical radioactive waste and for the thermal cutting of contaminated metallic components. In each of the above-mentioned cases, at JRC-Ispra, the activities are performed in a containment system (SAS confinement), equipped with forced ventilation and using specific personal protective equipment.
The application of the integrated ALARA approach to these activities has allowed the definition of appropriate reference levels for the relevant RP quantities.
For the waste treatment, the peculiar characteristics of the working environment and the operating mode taken into account led to the definition of specific levels for air concentration and surface contamination in the confinement. In the case of thermal cutting, acceptance criteria have been calculated in term of surface and mass contamination of the metallic component to be processed.

***Keywords**: ALARA, Treatment, Thermal Cutting*

PS3 (T3.3-0802)

Development of Radiation Protection Optimization Process to Calculate Dose Constraints

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Radiation protection optimization process is a source-related process, which considers economic and social factors to keep the individual doses as low as reasonably achievable. International organizations, including ICRP provided guidance on the radiation protection optimization process. Final objective of this study is to develop a systematic and practical process for radiation protection optimization. As a beginning step for the final objective, we investigated and analyzed radiation protection optimization process given by ICRP, IAEA, United States, United Kingdom, and Canada. ICRP proposed a 5-step process for optimizing radiation protection. The radiation protection optimization process is as follows: evaluation exposure situation, identification of protection options, selection of the best option, implement of the protection option, and evaluation of performance. ICRP also emphasized that the radiation protection optimization process should be continuously and cyclically implemented for the better radiation protection[1]. IAEA proposed a radiation protection optimization process similar to the ICRP[2]. United States, United Kingdom, and Canada applied their respective characteristics based on the ICRP optimization process. We developed a preliminary process for the radiation protection optimization based on the other cases. The developed optimization process consists of seven steps, which are given in Table 1. In this study, we reviewed other cases and developed a preliminary process for the radiation protection optimization. This study results will be used to develop a systematic and practical process for radiation protection optimization in Korea.

Table 1. Developed radiation protection optimization process

Step	Process	Content
1	Evaluation of exposure situation	· Evaluate exposure situation to clarify and identify the necessary factors for selecting protection options.
2	Identification of protection options	· Identify protection option to keep exposure as low as reasonably achievable.
3	Comparison of protection options	· Analyze each option considering distribution of individual exposure and identify the advantages and disadvantages of each option.
4	Sensitivity analysis	· Sensitivity analysis should be performed to increase the reliability of the results and to identify variables important to the optimization process.
5	Selection of best option	· Select best option considering qualitative and quantitative methods.
6	Implementation of protection option	· Implement selected protection option with clarifying responsibility.
7	Performance evaluation	· Evaluate performance with determining whether meet the objective criteria.

* It is necessary to repeat the process of evaluating the exposure situation if corrective options are needed

Keywords: Radiation protection, Optimization, Dose constraint

ACKNOWLEDGMENTS
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PS3 (T3.3-0860)

ALARA in focus

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By the 1950s the world had already realized the need to protect the people from the harmful effects of ionizing radiation. In 1955 ICRP established an incipient dose limitation system that has since been refined over the years. The development of the fundamental philosophy was completed in the ICRP Publication 26, in 1977, where the three principles of the dose limitation system were established and identified separately. The principles are known as: (a) justification, (b) optimization or (ALARA) As Low As Reasonably Achievable, and (c) dose limitation.

The philosophy contained in Publication 26 was adopted by the IAEA in its Safety Series publication No. 9, 1982 edition, and in Brazil by the National Nuclear Energy Commission in 1988, with the publication of the regulations CNEN-NE-3.01. The three principles remain valid and in the current editions of both documents.

This paper describes the development of the ALARA concept from its origin to the present day, to provide an insight and understanding of its meaning and importance for the new generations of Radiation Protection Supervisors and other professionals who work directly and indirectly in areas that involve the use of ionizing radiation.

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Poster Session T3

PS3 (T3.3-0867)

Preliminary External Dose Assessment of Workers in Radioactive Concrete Wastes Treating System for Kori-1 Decommissioning

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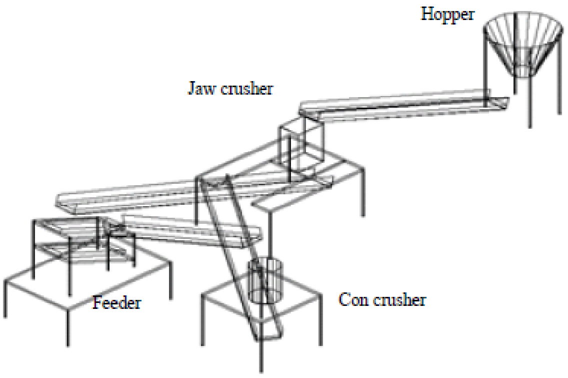
The waste treatment process is part of the decommissioning process of Kori-1 in Korea, which involves various processes such as crushing or screening the concrete. In this process, the worker may be exposed to the radioactive waste, so the analysis of how much dose is received during the work process is required[1]. In this study, the preliminary evaluation for radiation safety of the dismantled concrete treatment worker was conducted. VISIPLAN, a dose evaluation tool, was used to simulate the external exposure situation and the surrounding dose rate, and based on this, the safety of concrete waste treatment workers was evaluated.

The worker is exposed to only during inspection or failure because the process is a fully automated process. Therefore, the tasks and time needed to be inspected in the process of the processing system were derived. The treatment system was modeled using VISIPLAN and the results are shown in Figure 1. The contamination level of radioactive concrete was previously analyzed using the results of analyzing the radiation distribution of bio-shield using MCNP6 computing code, and only wastes with concentrations above clearance level were considered[2]. The radionuclides in the radioactive waste were considered ^{60}Co, ^{152}Eu and ^{154}Eu, which are the most radioactive nuclides in the bio-shield. As a result of the dose assessment, it was calculated that the assessment of the inspection scenarios would not exceed the annual dose limit, up to 0.3 mSv after all work. Based on these assessments, safety assessments in the treatment of other types of waste can be undertaken and contributed to the development of radioactive waste disposal worker guidelines.



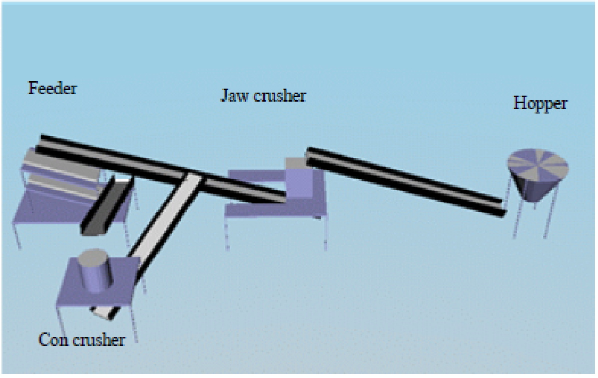


Fig. 1. Kori Unit 1 Radioactive Treatment System Process Concept Diagram

Keywords: Concrete, Occupational exposure, Treating system

ACKNOWLEDGMENTS

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PS3 (T3.3-1021)

Design of an Animal Irradiator Head for Minimizing the Radiation Leakage

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The linear accelerator (LINAC) based animal irradiator is being developed for various radiobiological researches at the Dongnam Institute of Radiological and Medical Sciences (DIRAMS). The animal irradiator is designed to emit 4 MV X-rays with a dose rate up to 3 Gy/min for the 15-cm diameter of field size at the 30-cm distance from the X-ray target. This type of LINAC consists of a C-band accelerator waveguide, microwave components, a high-power pulse modulator, and an irradiation head. Here, the irradiation head is comprised of various components such as X-ray target units, a primary collimator, a monitor chamber, and a secondary collimator. In terms of radiation protection, the shielding against the radiation leakage from the head components is a key factor. Also, the optimization of the head design is necessary to minimize the size and the cost of head housing. In this work, the radiation leakage was evaluated using the Monte Carlo N-Particle (MCNP) transport code according to sizes and depths of the beam hole in front of the target. In addition, the effect of external radiation shielding was evaluated. From the calculation results, the optimum geometry of the head components was determined at the dose for leakage radiations around the head to be within 0.5% of the primary radiation dose.

Keywords: *X-ray irradiator, Irradiator Head, Radiation Leakage*

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PS3 (T3.3-1024)

Shielding Evaluation for a 9-MeV Preclinical Research LINAC Facility

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In general, the preclinical studies include experimental researches with the irradiations of electron beams or X-rays to the cell lines or small animals to improve the radiotherapy technique. For the preclinical studies, a 9-MeV linear accelerator (LINAC) system is being developed and constructed in a shielding facility of the Dongnam Institute of Radiological and Medical Sciences (DIRAMS). Since the LINAC is a high-energy and high-dose-rate radiation generator, it is necessary to evaluate the radiation safety requirements for the operation. In this work, the radiation levels for primary, leakage, and scattered radiations at the interest areas around the operation room were calculated according to the National Council on Radiation Protection and Measurement (NCRP) recommendations. The workload (Gy/week) for X-ray experiments was determined for the calculations. In addition, the radiation leakages of the LINAC head were calculated with the emission angle of X-rays from the target by using the Monte Carlo N-Particle (MCNP) transport code. At last, we evaluated the regal requirements for our irradiation facility from the calculation results.

Keywords: *LINAC, Shielding evaluation, Radiation safety*

ACKNOWLEDGMENTS

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PS3 (T3.3-1078)

The Radiation Protection System Design of New Radwaste Treatment Facility

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The Korea Atomic Energy Research Institute (KAERI) has been prepared new Radwaste Treatment Facility to prevent recurrence since 2017. The new Radwaste Treatment Facility has a name as Storage and Treatment of All Radwaste (STAR) from KAERI. Conceptual design of the STAR facility was performed in 2017 and detailed design has been performing since 2018. In conceptual design, the STAR facility has an objective to treat and storage a very low-level waste (VLLW) and low-level waste (LLW) from KAERI. Therefore, the STAR facility can treat large amounts of radwaste produced from KAERI, which can make radiation-exposed to operators and worker in this facility. To prevent over-exposure to the worker, the radiation protection system was designed with these radiological characterizations of radwaste and ALARA (As Low As Reasonably Achievable) concept.

To evaluate the external exposure of worker of a nuclear facility, the annual permissible dose limit is to follow Korea regulation the 'standard for radiation protection, etc.' based on ICRP 60 recommendation: 5-year effective dose limit is 100 mSv (about 0.380 mSv/week), and annul maximum effective dose limit is 50 mSv. The performance of radiation protection system of the STAR facility was evaluated by using Microshield simulation program, and simulation was performed two cases: one is that work is a 1 m from the surface of the storage and the other is to add a 20 cm concrete wall and a 14 cm lead shield between the worker and surface of the storage: storage is selected due to having the highest radiation level in the facility. These simulation results are shown in Table 1. These simulation results showed that the dose of the worker who is protected by shielding structure is 0.0124 mSv/week, which is under 0.380 mSv/week. It means that this facility was designed enough to protect the worker and can secure worker safety from the radiation.

Table 1. The simulation results of external exposure for the worker

Case	Exposure dose
Worker	70.8 mSv/week
Worker (including the shielding structure)	0.0124 mSv/week

Keywords: *Radiation protection system design, exposure dose, worker safety*

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PS3 (T3.3-1129)

Preliminary Analysis for the PCS Neutron Monitoring System of Research Reactor

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The Reactor Protection System (RPS) of research reactor safely shuts down the reactor to protect the core by generating a trip signal to insert Control Absorber Rods (CARs) and Second Shutdown Rods (SSRs) into the core when the trip parameters exceed the predetermined setpoints. Among various instruments of RPS, Primary Cooling System (PCS) Neutron Monitoring System (PNMS) is installed to check a fuel failure by monitoring delayed neutron in the PCS of the research reactor. This paper proposes a calculation model and preliminary analysis results for the PNMS of KIJANG Research Reactor (KJRR). There are two kinds of neutron source in PCS: one is delayed neutron (DN) from fission product, the other is photoneutron (PN) by (γ, n) reactions with N-16. In this study, the DN is only considered as neutron source in PCS because the effect of photoneutron under normal operation was negligible from previous study [1]. The delayed neutrons in the primary coolant are due to a fission of the contaminated fissile material on the fuel surface, and then it is continuously measured under normal operation at the neutron detectors installed around the PCS pipe. If a fuel is failed during reactor operation and then the fission products from the failed fuel is mixed into the coolant, the PNMS detector shows a continuously increased count rate and reaches an unusually high count rate. Therefore, the key of PNMS analysis is to estimate the expected count rate of PNMS detector during normal operation. In order to estimate delayed neutron sources of KJRR by the fission products of uranium, the total amount of uranium contaminated on the fuel surface is calculated. In this process, it is assumed that PCS is a complete closed loop and all fission products from the contaminated uranium on the fuel surface have fallen into the coolant. Then, using the calculated amount of uranium, delayed neutron source term in PCS is evaluated and the count rate on the PNMS neutron detector is preliminarily evaluated. The neutron detector applied in MCNP modeling is a B-10 proportional detector, and it is simulated that three detectors are installed around the PCS pipe. As a result, the thermal fluxes at three neutron detectors are calculated as 2.22 neutrons/cm^2-sec, 2.26 neutrons/cm^2-sec, and 2.08 neutrons/cm^2-sec. Then, count rates at each detector are evaluated as 33.2 cps, 33.9 cps, and 31.1 cps, respectively. This value will be an important reference when the reactor trip setpoint of PNMS is decided.

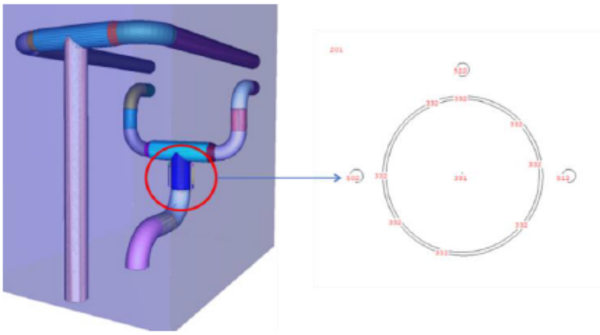

Fig. 1. MCNP modeling for PNMS

Keywords: *Detection of Failed Fuel, Delayed Neutron, PNMS*

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PS3 (T3.3-1205)

Review of Design Criteria of Auxiliary Facility in Access Control Zone

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In nuclear facilities, access control zones are designed to manage the radiation dose to radiation workers. In order to efficiently control the worker access to radiation control area, appropriate auxiliary facilities are required to access control zone. However, the design criteria for auxiliary facility varied by country. Therefore, it is necessary to review the design criteria by country and international organization. In this study, we investigated domestic design criteria of auxiliary facility in access control zone and compared the domestic criteria with the US and international organization criteria. The Korea Institute of Nuclear Safety (KINS) presented the design criteria of access control zone by publishing KINS/RS-N13.00 [1]. It recommended that access control zone should be located at the entrance of radiation control area and auxiliary facility should be designed in access control zone such as counting facility, decontamination facility, and change room etc. In United States, the Nuclear Regulatory Commission (NRC) presented the design criteria of access control zone by publishing Regulatory Guide 8.8 [2]. It provided a list of auxiliary facilities in access control zone and design features of each auxiliary facility. The International Atomic Energy Agency (IAEA) presented the design criteria of access control zone by publishing Safety Guide NS-G-1.13[3]. It provided a list of auxiliary facilities in access control zone. Table 1 shows a list of auxiliary facilities of access control zone in Korea, the US, and IAEA. The Korea and the US design criteria to auxiliary facility are at the same level. However, the IAEA recommended the additional auxiliary facilities such as health physics and first aid room. Therefore, at the stage of establishing general design criteria of domestic nuclear facility, these differences should be appropriately considered. The results of this study can be used as a basis for the design criteria of access control zone.

Table 1. Auxiliary facilities in access control zone

Korea (KINS/RS-N13.00)	U.S. (Regulatory Guide 8.8)	IAEA (Safety Guide NS-G-1.13)
▪ Counting facility ▪ Decontamination facility ▪ Change room ▪ Cleaning room ▪ Storage area for contaminated items	▪ Counting facility ▪ Personnel decontamination facility ▪ Change room ▪ Storage area for portable radiation survey equipment	▪ Health physics operations office ▪ Radiochemistry laboratory ▪ First aid room ▪ Personnel and equipment decontamination facility ▪ Change room ▪ Storage area for contaminated items and tools

Keywords: Access control zone, Auxiliary facility, Design criteria

ACKNOWLEDGMENTS

This work was supported through the KoFONS using the financial resource granted by NSSC. (No. 1805016)

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PS3 (T3.4-0980)

Dosimetric Impact of New ICRP Pediatric Mesh-type Reference Computational Phantom Series

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In 2016, the International Commission on Radiological Protection (ICRP) organized a Task Group (TG) 103 with the explicit purpose of developing new mesh-type reference computational phantoms (MRCPs) that can overcome limitations of previously used voxel-type reference computational phantoms (VRCPs) due to their finite voxel resolutions and the nature of voxel geometry. After completing the development of adult MRCPs, TG 103 is currently working on the development of pediatric MRCPs for five different ages (newborn, 1 year, 5 years, 10 years, and 15 years). The pediatric MRCPs, which is nearing completion, were constructed by converting the pediatric VRCPs into high-quality mesh format with preservation of the original topology of the pediatric VRCPs but with substantial improvements in detailed anatomy of some complex, small, or thin organs (e.g., skeletal tissues). In addition, the pediatric MRCPs include micron-scale radiosensitive target and source regions within the skin, lens, urinary bladder, respiratory tract organs, and alimentary tract organs. Note that these thin target and source regions could not be represented in the pediatric VRCPs. To investigate the dosimetric impact of new pediatric MRCPs, in the present study, the dose coefficients (DCs) of organ doses and effective doses were calculated for external exposures to broad parallel photon and electron beams by using the pediatric MRCPs. The results were then compared with those calculated with the pediatric VRCPs. The comparison showed that for photons, the organ dose DCs for most organs/tissues and the effective dose DCs of pediatric MRCPs were generally similar to those of pediatric VRCPs, except for very low energies (see figure 1(a)). For electrons, however, significant differences from the values of pediatric VRCPs were found, particularly for organ dose DCs of the superficial organs/tissues and skeletal tissues and also effective dose DCs (see figure 1(b)), which is because the representation of organs and tissues was improved in the pediatric MRCPs. The presentation will compare the organ dose and effective dose DCs for more exposure cases, including the internal exposure cases, and discuss the dosimetric impact of new pediatric MRCPs in more detail.

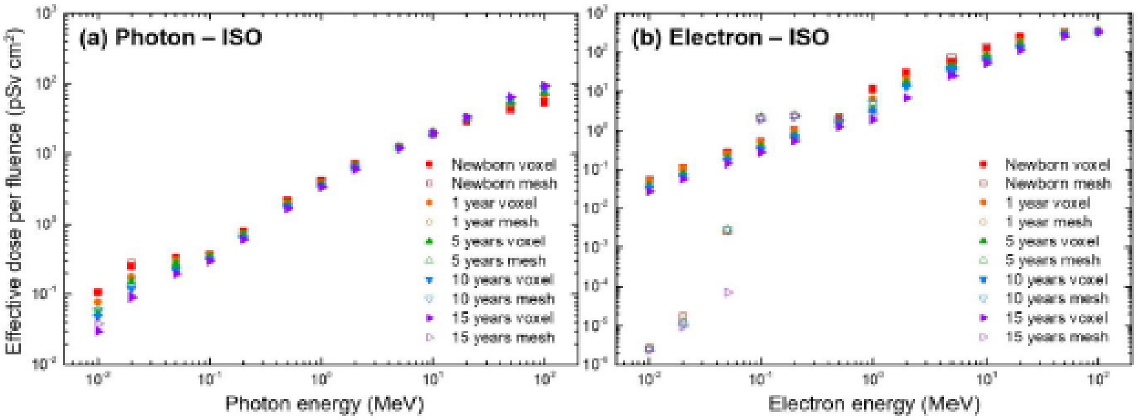

Fig. 1. Effective dose DCs of pediatric MRCPs along with those of pediatric VRCPs for (a) photons and (b) electrons in ISO geometry.

Keywords: ICRP, Pediatric mesh phantoms, dosimetric impact

PS3 (T3.4-0985)

Development of Adult and Pediatric Mesh-type Teeth Models to Calculate Enamel Dose Coefficients for Use in Retrospective Dosimetry

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Retrospective dosimetry is an essential tool for reconstructing or assessing the doses to individuals exposed to ionizing radiation from radiological accidents. Electron paramagnetic resonance (EPR) dosimetry with tooth enamel is one of the most widely used physical method to reconstruct radiation doses in the field of retrospective dosimetry [1]. The assessed enamel absorbed doses can be used to estimate the radiation risk by converting them into organ absorbed or effective doses. Several research groups used adult mathematical and/or voxel phantoms to calculate dose conversion coefficients by performing Monte Carlo simulations. However, due to the simplicity of mathematical phantoms and limited resolutions of voxel phantoms, the complex structure of tooth enamel was not fully defined in these phantoms. In addition, organ absorbed or effective doses for pediatrics could not be accurately estimated with the existing dose conversion coefficients for adult. To overcome these limitations, in the present study, the mesh-type detailed teeth models, which realistically define complex tooth enamel structure, were developed and incorporated into the adult (male and female) and pediatric (newborn, 1-, 5-, 10-, and 15-year-old male and female) ICRP mesh-type reference computational phantoms (MRCPs) to establish a dataset of enamel dose coefficients. To calculate site-specific enamel dose coefficients, each tooth was modeled separately considering the location and eruption period of each age [2]. For this, the masses of tooth tissues (i.e., enamel, dentin, pulp, and cementum) for both the erupted and unerupted tooth were calculated while matching the total teeth mass to the reference value for all ages [3-5]. Note that the high-quality PM models for permanent and deciduous teeth were employed to model the outer surface of each tooth; each tooth was scaled to match the reference mass and placed in the head of adult and pediatric MRCPs. Then, the tooth tissues were defined in each tooth referring to the calculated tissue mass data. After constructing the mesh-type detailed teeth models, the enamel dose coefficients were calculated for mono-energetic photon beams in idealized irradiation geometry (antero-posterior (AP), postero-anterior (PA), left-lateral (LLAT), right-lateral (RLAT), rotational (ROT), and isotropic (ISO)) ranging from 10 keV to 10 MeV by performing the Geant4 Monte Carlo radiation transport simulations. The presentation will describe the overall development procedure of the mesh-type teeth models in more detail and discuss the calculated enamel dose coefficients compared with those of the previous researches.

Keywords: Enamel dose coefficient, Mesh-type computational phantom, Monte Carlo

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PS3 (T3.5-0038)

Establishment of Reference Levels in Activity and Risk Assessment of Radiation-Induced Cancer in Nuclear Medicine Practices

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The International Commission on Radiographic Protection (ICRP) has emphasized the importance of accurately determining the average dose levels, or administered activity, received by patients in each medical procedure that makes use of ionizing radiation in that context the nirea software was developed to obtain the Reference Levels in Activity (NRA) for nuclear medicine patients. The program also allows obtaining dose values absorbed in critical organs based on patient specificities, age, sex and Body Mass Index (BMI) in order to evaluate the risk involved in each exam. The objective of this work is to study the through estimates of doses absorbed in critical organs obtained by the NIREA software in order to determine factors of possible increases in the risk of developing secondary cancer. The risk assessment methodology provided by the BEIR VII report was used for this carcinogenic risk estimate. Used the mathematical models of the BEIR VII was able to estimate the risk attributable over the lifetime LAR.

Keywords: *Carcinogenic Risk Estimate, Reference Levels in Activity, BEIR VII*

ACKNOWLEDGMENTS

The authors would like to thank Comission National of Nuclear Energy

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PS3 (T3.5-1008)

Korea's Nuclear Safety Law Issues Revised after the Fukushima NPPs Accident

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Article 1 of the Atomic Energy Safety Act (formerly, the Atomic Energy Act) enacted in 1958, "prescribes safety management related to the research, development, production, and use of nuclear power, aiming at the prevention of disasters caused by radiation and the public safety." In order to achieve the purpose of these laws, site expertise and effectiveness must be guaranteed. However, since the Fukushima accident in 2011, the Korean Nuclear Safety Act has undergone rapid revisions without full consideration of site conditions, and voices of complaints and anxieties about the organizational and social variables that can cause human error, have emerged. The administration should consider the theory that "safety accidents are caused by a combination of errors in the safety of the dangerous goods themselves, the safety of the facility, the organizational safety system, and human safety attitudes." Otherwise, the predictable safety accident rate will increase. The process of exposing risk variables and reducing their risk factors is to prevent accidents. However, the Nuclear Safety Law, which has been revised in Korea after the Fukushima accident, is not focused on the effective management by radiation safety managers on site due to excessive regulations. The actual safety management of the workers has become difficult, and the reality is that formal procedure-oriented paperwork must be focused on. This is because the implementation of penalties against site workers has generated behavioral restrictions, and created stress for NPP workers. Cases of administrative disposal at sites in 2017-2018 have shown to enforce a 100 million won penalty at specific sites. However, the penalties which have been enforced are only the result of non-compliance with some safety management regulations rather than actual accidents which have occurred at NPP's. At present, Korea's nuclear safety laws are aimed at strengthening safety regulations, however this strengthening is not enhancing overall safety at Korean NPP's. If regulations are unrealistically strengthened, site safety will only follow in certain circumstances. The current law states that radiation safety managers can be dismissed if they neglect their work. There is no provision in the Nuclear Safety Act regarding a worker's inefficient or "lazy work habits." The government's interpretation of these departments is to exercise great discretion. If so, there is a compelling discussion about administrative expertise and coherence of regulations. Among the nation's efforts to protect the people, the most important thing is to protect the "basic human rights" of the people. Therefore, in order to secure public safety, the government must first protect safety management on NPP's. Since the safety management of the site is in the hands of the on-site experts, the national safety design, including the on-site experts, should be addressed. The national government should provide sufficient conditions for the advancement of technology and skills. If there are no field-based laws, no conditions for strengthening safety, and unilateral regulations, public safety may be violated. This may result in administrative plans and guidance that violate the Constitution. The legislature, the executive, and the judiciary should also consider whether the Nuclear Safety Act is regulated to protect radiation for the state and the people. The purpose of this study is to find the excessive provisions of the Nuclear Safety Act, which was radically revised after the Fukushima nuclear power plant accident. In particular, this paper analyzes the revised regulations that are considered excessive in safety compliance given the reality of each situation or circumstance.

Keywords: Atomic Energy Safety Act, Radiation Safety, Amended Regulation

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PS3 (T3.5-1188)

Radon Concentration in Radiation Areas of Primary, Secondary and Tertiary Medical Institutions at South Korea

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This study aims to measure the Indoor Radon Concentration (IRC) of 56 radiation areas of 34 primary, secondary, and tertiary medical institutions to assess their adequacy as required by the Ministry of Environment (MOE), Environment Protection Agency (EPA), and International Commission on Radiological Protection (ICRP).

The study was conducted from July 2019 to February 2020, and measurements were obtained using FRD400S. Primary, secondary, and tertiary medical institutions were included, with 15 radiation areas (15 medical institutions), 18 radiation areas (6 medical institutions), and 23 radiation areas (12 medical institutions), respectively. Statistical analysis of each medical institution was carried out using Games-Howell post-hoc test for Welch's one-way analysis of variance.

The Avg-IRC for primary, secondary, and tertiary medical institutions were 81.39 (median : 100.33) Bq/m^3, 40.34 (median : 37.63) Bq/m^3 and 36.3 (median : 35.05) Bq/m^3, respectively. There was a statistically significant difference between primary and tertiary (p=0.001, <0.01) and between primary and secondary (p=0.002, <0.01). However, there was no statistically significant difference between secondary and tertiary (p=0.404, >0.05).

The IRC of eight radiation areas has exceeded the ICRP's recommendation is located on the primary medical institutions. The max-IRC and min-IRC showed an 83.36% difference, with actual value of 131.16 Bq/m^3 and 21.82 Bq/m^3, respectively. Therefore, radiation areas located on the primary medical institutions must set up preventative measures to minimize exposure, as outlined by ALARA principle.

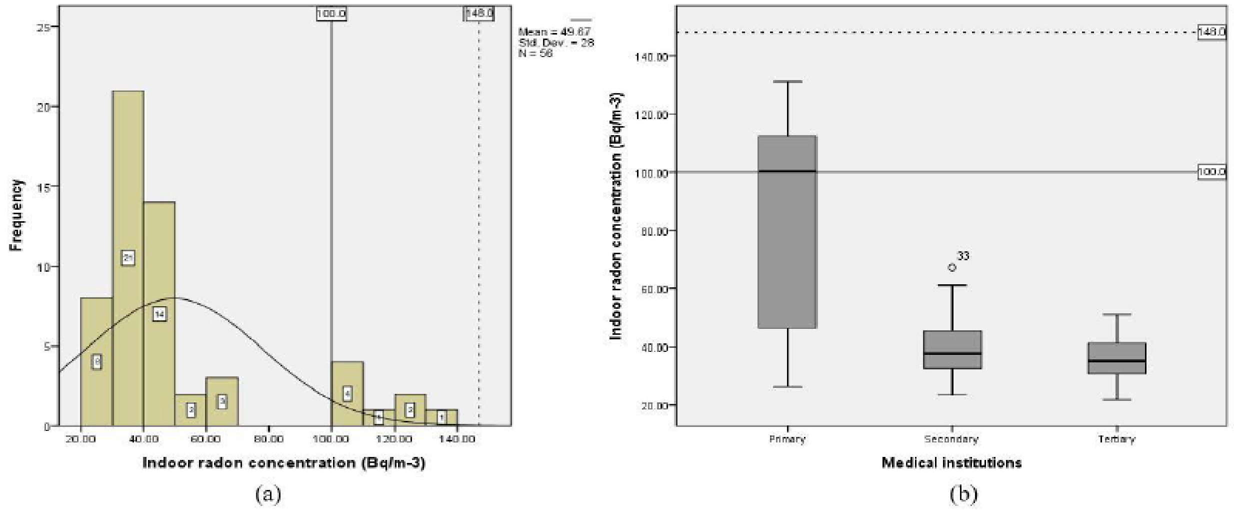

Fig. 1. Distributions of IRCs in comparison to two lines of solid and dot, guidelines for ICRP and MOE, EPA, respectively. (a) Distribution of 56 radiation areas in all medical institutions. (b) Distribution of IRCs at primary, secondary, and tertiary medical institutions

***Keywords**: Indoor Radon Concentration, Radiation Area, Medical Institution*

PS3 (T3.6-0493)

Hydrogeochemical And Environmental Isotope Characteristics Of The Thyspunt Area, Eastern Cape, South Africa

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The Thyspunt area, located in the Eastern Cape South Africa is a prospective nuclear site and like most areas in the Eastern Cape, it is heavily reliant on groundwater resources. Both surface and groundwater resources are potentially vulnerable to contamination and pollution emanating from anthropogenic activities such as nuclear power plants. Therefore, the understanding of the hydrogeological dynamics of the Thyspunt nuclear site is paramount as it serves as a basis for policy and decision making pertaining to the protection of the environment. The purpose of this work was to understand the hydrogeological conditions of the area through the application of environmental isotopes and hydrogeochemical analyses. The hydrogeochemical and environmental isotope signatures were used to identify the interaction between various water bodies, major hydrogeochemical processes, and possible sources of moisture in the Thyspunt area. Fifty-nine water samples from springs, wells, streams and the ocean were collected for major ions, metals, and environmental isotope analyses. The groundwater is characterised by electrical conductivity that varies between 286 and 7040 μS/cm, dominant alkaline pH conditions and calcium-magnesium-bicarbonate hydrochemical water type. Hydrochemical evolution of groundwater is observed along the groundwater flow direction (west to east), from fresh calcium-magnesium-bicarbonate water type to saline sodium-chloride water type. Furthermore, mixing of calcium-magnesium-bicarbonate and sodium-chloride type groundwater is apparent in the analysed spring samples, indicating deep groundwater circulation. Gibbs plot of major ion hydrochemical data indicates that the groundwater hydrochemistry is primarily controlled by water-rock interactions (mineral dissolution) and evaporation processes. Isotopically, water samples from springs and wells have depleted isotope signatures, indicating rainfall recharge from either high altitude moisture source or recharge during colder seasons or both. The lack of isotopic signature similarity between ocean

Poster Session T3

PS3 (T3.6-0493)

Hydrogeochemical And Environmental Isotope Characteristics Of The Thyspunt Area, Eastern Cape, South Africa

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water and groundwater signifies the absence of seawater intrusion in the sampled aquifers. Deuterium excess values range between -0.71 ‰ and 22.64 ‰, suggesting the presence of numerous moisture sources. Tritium activity in groundwater varies between 0.2 T.U and 3.2 T.U, showing submodern to modern (5 – 10 years) recharge. Hydrochemical and environmental isotope similarities between spring and borehole samples confirm the fact that springs are a surface manifestation of the local groundwater flow conditions.

Keywords: Environmental isotopes, hydrogeochemistry, seawater intrusion, hydrochemical evolution, regional recharge and hydraulic link.

PS3 (T3.6-1083)

A Review of Environmental Impact Statements on Dismantling of Power Plants

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Kori Unit 1 and Wolseong Unit 1 are being prepared for dismantling in Korea. There are needs to consider the media of radiological environmental impact assessment that should be taken care of during the preparation for decommissioning. In this paper we will glance at some cases that have already completed dismantling overseas. The United States regulates decommissioning under the NRC, and nuclear reactor license holders are required to submit a report called PSDAR within two years of permanent shutdown. The report then compares and identifies existing environmental impact assessments (during the operation) and environmental impacts for planned decommissioning activities. The United States will confirm the conformity with the results of general EIS's environmental impact assessments regarding nuclear decommissioning activities and, if necessary, evaluate the site characteristics. Literature review of US decommissioning cases have shown that all US decommissioning plants have satisfied general EIS and therefore do not require additional site specific environmental impact assessments. In Korea, the environmental impact assessments is carried out in the periodic safety assessments during the operation of nuclear power plants, and the final decommissioning plan, which is currently under development, also describes the status and monitoring of the site environment. This review is expected to help establish an environmental monitoring plan for decommissioning.

Table. 1. Summary of the Environmental Impacts from Decommissioning Nuclear Power Facilities [1]

Issue	Generic	Impact
Onsite/Offsite Land Use		
- Onsite land use activities	Yes	Small
- Offsite land use activities	No	Site-specific
Water Use	Yes	Small
Water Quality		
- Surface water	Yes	Small
- Ground water	Yes	Small
Air Quality	Yes	Small
Aquatic Ecology		
- Activities within the operational area	Yes	Small
- Activities beyond the operational area	No	Site-specific
Terrestrial Ecology		
- Activities within the operational area	Yes	Small
- Activities beyond the operational area	No	Site-specific
Radiological		
- Activities resulting in occupational dose to workers	Yes	Small
- Activities resulting in dose to the public	Yes	Small
Radiological Accident	Yes	Small
Noise	Yes	Small
Transportation	Yes	Small

Keywords: Environmental Impact Statement, Decommissioning preparation period

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Poster Session T3

PS3 (T3.7-0088)

Current status of Radiopharmaceuticals production in Brazil: Licensing and radioprotection aspects

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In Brazil, as has been occurring worldwide, the number of procedures using radiopharmaceuticals is increasing. The production and selling of short half-life radioisotopes used to be monopoly of the Brazilian Government. In 2006, a Constitutional Amendment revoked the state monopoly due to the need for the use of short half-life radioisotopes in nuclear medicine centers very far from the government production facilities. The aim of this study is to describe the current status of radioisotopes production and sales in Brazil and discuss some licensing process. Currently there are 14 radiopharceuticals production facilities and 4 radiopharmacies operating in Brazil. The type of licensing process conducted in Brazil does not take into account the population density of each state, with a free competition model being adopted. Because of this there is a lot of equipment concentrated in the Southeast and no cyclotrons or radiopharmacy operating in the Northern part of the country. One of the biggest obstacles during the licensing process is the designation of qualified personnel in radiopharmacy or accelerator for radiopharmaceutical production as operation workers and radiation safety officers. Currently there are only 17 qualified workers in these fields. Regarding regulatory inspection in Brazil, during the facilities licensing process two types of inspections are usually performed: one to monitor the radiopharmaceutical production (usually overnight) and another to verify records and to test security systems. The number of facilities for radiopharmaceuticals production and sales are increasing. However, a number of external factors such as the distance from the nuclear medicine centers, and qualified personnel have proved crucial for the economic viability of this type of facility, and a rigorous licensing process is necessary to ensure radiological protection

PS3 (T3.7-0191)

Performance Evaluation of Multi-Array Plastic Scintillation Detector Using Static and Dynamic Source Conditions

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Radiation portal monitor (RPM) systems applying plastic scintillator of large-size have been deployed to monitor the illegal radionuclides at border crossings world widely. Considering the low collection efficiency of the optical photons for conventional plastic scintillator, increased photo-multiplier tubes (PMTs) can improve detection efficiency. Therefore, in this study, a performance of the multi-array plastic scintillator was evaluated and its validation for radionuclide identification was conducted in various measurement condition.

14 hexagonal plastic scintillators of 24 cm diameter were combined as one detector, and each scintillator were individually bonded with PMT. And energy spectra of ^{137}Cs (8.6, 74.6 μCi), ^{60}Co (6.6, 13.2 μCi), ^{226}Ra of 14.4 μCi and ^{40}K in KCl of 200 kg were measured in static and dynamic condition with vehicle, and evaluated with the energy weighted algorithm previously proposed [1] and convolutional neural network (CNN) method [2].

Comparing with conventional scintillator, the multi-array scintillator measured 18.6% and 16.9% increased count in energy spectra as Fig.1 for ^{137}Cs (in 0.44-0.52 MeV) and ^{60}Co (in 0.97-1.05 MeV), respectively. Through the CNN method, ^{137}Cs and ^{60}Co were discriminated over 90%. Also ^{226}Ra and ^{40}K which have similar theoretical Compton edge with ^{137}Cs and ^{60}Co were identified over 70% in dynamic source condition.

In this study, the performance of multi-array plastic scintillator showed higher detection efficiency for Compton maximum area. In addition, the four radioactive sources were successfully identified using the machine learning algorithm. We expect that the type of this system could contribute to reduce nuisance alarm at RPM deployment site by effectively preventing the illicit trafficking radioactive materials.

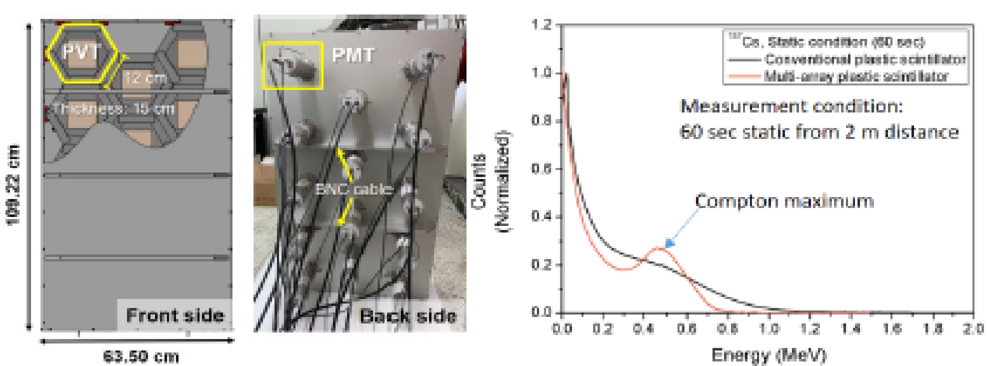

Fig. 1. Multi-array detector composed of 14 plastic scintillators (left) and energy spectra of ^{137}Cs measured by conventional and multi-array plastic scintillator (right).

Keywords: Radiation portal monitor, Multi-array plastic scintillator, Radioisotope identification algorithm, Convolutional neural network

ACKNOWLEDGMENTS

This work was supported by the Radiation Technology R&D program through the National Research Foundation of Korea; the Ministry of Science and ICT (grant number 2017M2A2A4A01019817).

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PS3 (T3.7-0681)

A Graphical Approach for Constructing System Materials for H-RMC used in MLEM-based Image Reconstruction

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The Hemispherical Rotational Modulation Collimator (H-RMC) is a radiation imaging system developed as an attempt to overcome the drawback of the conventional Rotational Modulation Collimator (RMC) system, the limited field-of-view (FOV). It can collect radiations from wider directional range (2π) than conventional one does. The system obtains the modulation pattern by placing the radiation detector at the center of the hemispherical collimator set, consisting of inner and our collimators. By rotating the collimator set, one can obtain a variation of the count value corresponding to the open area of the collimator set and the incident radiation flux. In the previous study [1], the validity of H-RMC was studied with Monte Carlo simulations using MCNP6 was verified.

In this study, we develop a graphical approach to quickly generate the modulation curve and the system matrix of the H-RMC to be used for image reconstruction, using a Matlab® simulation. This analytic approach creates deterministic results for the modulation pattern for each source position. The design of H-RMC rotated by each respective angle was sequentially imported to a digital image shown from a certain source position with the bird's eye view. The number of pixels of color corresponding to the detector shown through the collimators were counted for each image to obtain the modulation pattern. Based on the previously studied H-RMC design [2], we created the graphic image and calculated modulation patterns for the source located at θ = 30° and 60°. The results were compared with simulated modulation patterns obtained by MCNP6 as shown in Figure 1. Compared to the Monte Carlo simulation, this method could obtain the whole modulation pattern for each source position in a significantly reduced amount of time, which is about 2 hours with a single thread calculation with Matlab®. The same simulation with MCNP6 normally took ~24 hours. Both results show some discrepancy between each other, and we speculate that the difference can be adjusted by further considering physical factors such as the intrinsic detector response and the shielding effect of the collimator. We will show gradual enhancement of the reconstructed image implementing MLEM (Maximum Likelihood Expectation Maximization) method, as we include additional factors to correct the results obtained by the graphical approach.

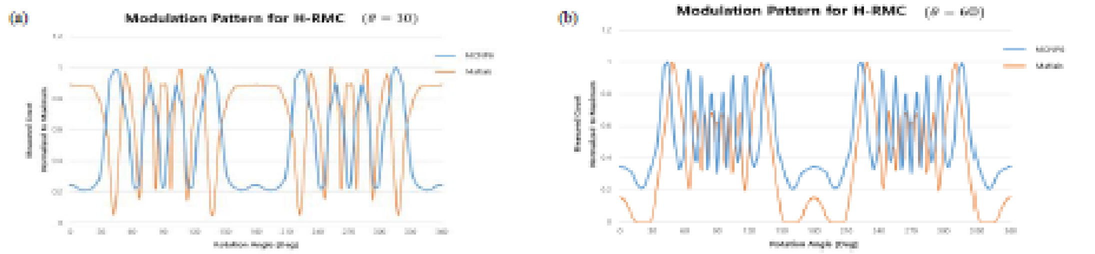

Fig. 1. Modulation patterns obtained by MCNP6 and graphical approach for the source located at (a) 30° and (b) 60°.

Keywords: H-RMC, Modulation Pattern, Graphical approach

ACKNOWLEDGMENTS

This work was supported by the Korea Institute of Energy Technology Evaluation and Planning(KETEP). (No. 20181520302230)

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PS3 (T3.7-0833)

The combination of pterostilbene, silibinin, nicotinamide riboside and FSL-1 lipopeptide exerts high protection against radiation injury in mice

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Electromagnetic radiation, including X-rays and γ rays, damages cells mainly by direct ionization of DNA and other cellular targets and by indirect effects through reactive oxygen species. The response to radiation exposure depends on the cell type and dose of radiation, inherent tissue sensitivity and repair, and modulating intracellular factors that include cell cycle status, O_2 pressure, and levels of thiols and other antioxidants.

The development of protective agents against exposure to harmful radiations has been a subject of intense investigation for decades. However, no ideal, safe radioprotectors are available. To date, no drug has been approved for the treatment of exposures to high levels of ionizing radiation that lead to the development of hematopoietic and gastrointestinal acute radiation syndromes (ARS).

A wide range of phytochemicals (compounds produced by plants, generally to help them thrive or thwart competitors, predators, or pathogens) are antioxidants and, thus, potential radioprotectors. We reported that topical administration of Pterostilbene (3,5-dimethoxy-4'-hydroxystilbene, PTER, a natural stilbene originally isolated from the heartwood of red sandalwood) protects against chronic UVB radiation-induced skin carcinogenesis, and protects the skin against the burning and irritation effects of acute UVB doses (1). The protection elicited by PTER is not due to a physico-chemical screen or to direct antioxidant effects. PTER induces an increase in our physiological antioxidant defences. Therefore, it is plausible that PTER, if administered systemically (IV or IP), could also exert protection against other types of ionizing radiations.

We investigated the effect of PTER, and found that it decreases the γ-radiation induced mortality (95% in controls vs 60% in PTER-treated mice at day 60 post-irradiation) in Swiss albino mice (male, 9-10 weeks old, 30-32 g) which received a LD50/30 total-body γ irradiation. However, the combination of PTER and silibinin [SIL,(2R,3R)-3,5,7-trihydroxy-2-[(2R,3R)-3-(4-hydroxy-3-methoxyphenyl)-2-(hydroxymethyl)-2,3-dihydro-1,4-benzodioxin-6-yl]-2,3-dihydrochromen-4-one, a natural flavanone isolated from the silybum marianum (milk thistle)] further reduced mortality (25% at day 60 post-irradiation). These results prompted us to investigate if addition of potential radiomitigators could improve the protection against radiation-induced injury.

NAD+ is an essential coenzyme in ATP production, and NAD+ consuming enzymes like poly(ADP-ribose) polymerases and sirtuins are important regulators involved in chromatin-restructuring processes during repair and epigenetic/transcriptional adaptions. The effect of nicotinamide, a precursor for NAD+ synthesis, on the DNA repair capacity following γ and UV irradiations was originally studied in several repair-proficient and repair-deficient cell lines. Nicotinamide was shown to improve the repair capacity in a concentration-dependent manner (2). Therefore, NAD+ precursors, if efficient under *in vivo* conditions, could be potential radiomitigators. Besides, replenishment of hematopoietic sites is critical for recovery following radiation exposure. Interestingly, a recent report shows that the Toll–like receptor (TLR) 2/6 agonist, FSL (fibroblast-stimulating lipopeptide)–1 (FSL-1), therapeutically mitigates ARS (3). We found that addition to the treatment combination of nicotinamide riboside (NR, a NAD+ precursor) and FSL-1 promotes long-term survival in mice (90% one year after irradiation).

Keywords: Polyphenols, Nicotinamide Riboside, FSL-1

ACKNOWLEDGMENTS

This work was funded by the University of Valencia (Spain) and Elysium Health Inc. (NY, USA)

PS3 (T3.7-0877)

An urgent need for the simplification of the present system of radiation protection quantities in applications of radiation and nuclear technologies

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During its more than one hundred long history, the system of the quantification of radiation exposure underwent through a number of phases, where many new elaborated quantities reflecting understanding and the level of knowledge of radiation effects at that time were applied. As it happened, newly introduced quantities, which were usually defined in an even more complex manner than previous ones, were always for some time used together with the older quantities. Moreover, in the definition of novel quantities emphasis was paid much to their too strict relation in reflecting biological effects and less attention to the problems of their direct measurement and monitoring. This is why we have now in use quantities recommended for the regulatory control of radiation exposure, which cannot be readily experimentally and reliably assessed in practice. Moreover, there are too many quantities currently used in radiation protection where there is also some confusion in relevant units where quite often the unit Sv is applied often to quantify high exposures characterized by deterministic rather than stochastic effects. The paper presents an overview of the existing situation in this field and proposes some changes. It is suggested to limit the use the current, rather complicated system of radiation protection quantities by scientists at universities and research institutes, while for routine monitoring to develop and adopt a much simpler system based on measurable quantities. Anyway, one has to admit that the present system of quantities is so intricate that even radiation workers engaged in routine use of radiation and nuclear technologies cannot fully appreciate and implement it, including correct interpretation of the results of their measurement and monitoring.

PS3 (T3.7-1204)

Assessment of Radiation Dose Assuming a Research Reactor Accident Based on Representative Person Concept

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When an accident occurs during operating a nuclear power plant, it results in radiation exposure to residents around nuclear power plant. Therefore, the Nuclear Safety and Security Commission (NSSC) sets notice that protect residents from nuclear power plant accident[1]. The International Commission on Radiological Protection (ICRP) recommended applying the representative person concept to reasonably assess radiation dose to public and it expected to be reflected in domestic regulation[2]. The objective of this study was to assess radiation dose assuming a research reactor accident using representative person concept. In order to define source term, accident should be defined. The biggest radiological accident occurring in a research reactor is Channel Flow Blockage accident[3]. Therefore, we assumed that there was a Channel Flow Blockage accident at a research reactor. Also, in order to assess radiation dose applying representative person concept, exposure scenarios and exposure pathways should be defined. Therefore, three exposure scenarios were set up: (1) adult residents, (2) 10-year residents and (3) 1-year residents. In the case of exposure pathway, two exposure pathways that residents could be exposed at the early of accident were set up: (1) external exposure from radioactive cloud and (2) internal exposure from inhalation. In the case of atmospheric dispersion factor, it should be considered the actual reside point value to assess radiation dose applying representative person concept. Using PAVAN program, we evaluated the actual reside point atmospheric dispersion factor. In the case of breathing rate, we considered Reg.Guide 1.4 value. Also for conservative dose assessment, we assess 7 days cumulative dose. Table 1 shows the results of dose assessment by exposure scenarios and exposure pathways. The concept of representative person was applied to assess the resident around research reactor accident. Radiation dose was 5.34 mSv for adult residents, 7.28 mSv for 10-year residents and 6.36 mSv for 1-year residents. Therefore, the radiation dose to representative person was 7.28 mSv for the 10-year residents. The result of this study can be used as a prior study to accept the concept of representative person recommended by ICRP 103 in Korea in the near future.

Table 1. Representative person dose assessment results (mSv)

Exposure Pathway	External exposure	Internal exposure	Total
Adult Residents	4.24	1.10	5.34
10-year Residents	4.72	2.56	7.28
1-year Residents	5.10	1.26	6.36

Keywords: Representative Person, Dose Assessment, Accident

ACKNOWLEDGMENTS
This work was supported through the KoFONS using the financial resource granted by NSSC. (No. 1805016)

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PS3 (T3.8-0222)

How international information and comprehensive safety assessment can be helpful to lessen radiological incidents or accidents

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As time goes by and experience is gained, modernization and technological development show the need to implement more complex programs and procedures to prevent the occurrence of new events involving radioactive material, particularly in industrial and medical applications. In the last decades there have been numerous efforts and contributions to improve and integrate various issues related to radiation safety, to avoid radiological events. This effort needs to continue and facilitate the inclusion of new approaches, as safety assessment, developing and implementing different mechanisms and strategies to strengthen capacity building, to lessen the occurrence of radiological events. Nevertheless, the evaluation of international information concerning the occurrence of radiological events, show that even after this effort, new events keep occurring weekly. As examples, we have the International Nuclear Event Scale (INES) as a communication tool about safety significance and Safety in Radiation Oncology (SAFRON) for medical purposes. As an example, a large list of events involving industrial radiography and the loss of control of radioactive material are informed around the world, where a list of anomalies, failures, incidents and near misses are detected, and in many cases, the events reported are similar or very similar to other occurred in the past. The information of radiological events occurring around the world is taken in some cases as news, but without giving it the potential value that corresponds to it, and in general is not used to improve changes or better measures or procedures that can prevent similar radiological events in the future. According to the IAEA GSR Part 3, Part 4.1 and Part 7, recommendations on responsibilities for protection and safety, assessment of occupational exposure, emergency management system, safety assessment, prevention and mitigation of accidents, cooperation and procedures must be applied to prevent risky situations. This is the reason why in this paper, will be described the mechanism to implement a global action plan, creating an evaluation committee in the regulatory body or competent authority, to analyze different events occurred in the past and the present, with the main objective of prepare a sort of recommendations and rules to prevent more radiological events. The inclusion of this initiative will offer a change in risk assessment concept regarding the use of lessons learned and not learned and experience gained. As a result of the evaluation provided by committees, recommendations can be integrated and used by different countries, generating an exchange information strategy, that lead to implement better safety assessment. This action plan require two steps: the evaluation of events, vulnerability, risk communication, change attitude in the prevention of risks, accident rate, human factors, weaknesses of the regulatory system, different scenarios evaluation and common cause failure, and, the technical application of the result of this recommendations introducing more deep safety assessment, offering a comprehensive radiological risk management. An action towards a solution must be the use of reported events as an information tool to improve the reaction of regulatory bodies or competent authorities to prevent new radiological events. At the same time to provide means for Member States to establish an effective safety assessment applying the result of this initiative to promote and enhance safety globally.

Keywords: safety assessment, radioactive material, radiological accidents

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PS3 (T3.8-0237)

The formation of human resources in radiation protection and the safety of radiation sources in Brazil: the postgraduate educational course (PGEC) of IRD/IAEA

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International Congress Programme Committee of IRPA15 cordially invites you to submit an abstract to the 15th International Congress of the International Radiation Protection Association (IRPA) to be held on 11~15 May 2020 in Seoul, Korea. All abstracts must be submitted electronically through the website only. Abstracts submitted via e-mail, fax or regular mail will neither be accepted nor acknowledged. Submitted abstracts can be revised on the website during the abstract submission period. All submitted abstracts will be reviewed and assigned to appropriate session. Notification on acceptance will be sent to the submitter by email on 31 December, 2019 for oral presentation and 31 January, 2020 for poster presentation. The Postgraduate Course in Radiation Protection and Security of Radioactive Sources (PGEC) [1], offered by the Institute of Radiation Protection and Dosimetry (IRD) in partnership with the International Atomic Energy Agency (IAEA) was designed to meet the needs of professionals who are working in the field of radiation protection and safety. Created in 2011, provides the foundation tools necessary for those who will become instructors in their area, forming qualified experts so that will act as multipliers of the knowledge of the area. The IRD [2] is a Brazilian institute for nuclear research, with the objective of acting as a national reference center in the areas of radioprotection, dosimetry and metrology of ionizing radiation, AND work in collaboration with universities, government agencies and industries. Offers regular education and training courses, and is collaborating with the IAEA as a Regional Training Center (RTC). The course has a workload of 472 hours, divided into 17 modules, and lasts 26 weeks, with theoretical parts and practical training. Some theoretical topics and exercises are developed online using the courses virtual classroom. The course structure also takes into account the requirements of the "International Basic Safety Standards for Protection against Radiation Sources (BSS)", "IAEA Safety Series No. 115 (1996)" and related safety recommendations on "Safety Guides. At the end of the course students must submit a project, called Final Work (FW) to the Graduate Committee to complete the minimum requirements to obtain their certificate. The teachers (professors and researchers) are composed mostly of staff from the IRD.

Keywords: Postgraduate educational course (PGEC), Radiation Protection, Brazil

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PS3 (T3.8-0515)

Awareness of Clinical Imaging Guidelines use among referrers in developing countries; a case of Mengo Hospital, Uganda

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Background; Inappropriate imaging is risky if ionising radiation is used. Clinical Imaging Guidelines (CIGs) have been developed and used in developed countries to reduce inappropriate imaging. Knowledge and awareness of use of such diagnostic decision-making tools in the developing world like Uganda, hasn't been assessed. **Objectives**; To determine the awareness level of CIGs use among referrers of head CT exams. **Methods**; A clinical audit using a structured questionnaire was conducted among 21 head CT referrers at Mengo Hospital. Assessment was done pre and post intervention (training on radiation protection and CIGs use). The questionnaire was divided into 3 themes namely: definition of CIGs-4 questions, use of CIGs-11 questions and radiation protection-3 questions. The number of correct responses scored by participants was presented as a proportion. A percentage difference between the pre and post assessment for each theme was determined to indicate change in awareness levels. **Results;** Our baseline results showed that 46% of the referrers could correctly define CIGs, 61% had knowledge regarding use while 41% could relate CIGs to radiation protection. We also found that after the intervention 79% of referrers could correctly define CIGs, 94% had knowledge regarding use while 83% could relate CIGs to radiation protection. The percentage increase in CIGs awareness based on correctly defining, use and relating them to radiation protection was 26.4%, 21% and 34% respectively. Training on CIGs use should be introduced to improve knowledge and awareness.

Key words; Radiation protection, clinical imaging guidelines, CT

PS3 (T3.8-0604)

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In 2019, was created the UY-105 rule by the national regulatory authority on radioprotection (ARNR) which regulates the presence of radiation protection officer in medical and industrial areas where it works with ionizing radiation. In Uruguay there is a lack of radiation protection officer, in order to correct this situation, the Academic Unit of Radioprotection and the ARNR in a technical cooperation project with the IAEA (URU-9011) chaired a workgroup whose task is to design the first RPO course. The workgroup with some IAEA experts, began to develop a graded approach to the country's needs. A document was produced presenting training needs, the design of the national education and training program. The master program includes topics includes in Syllabus Training RPO. The structure of the course follows the recommendations made by OIEA experts during their visits in May and September 2019.The aim of the course is to prepare highly qualified staff in order to be able to perform in several medical spheres. The admission to this course requires a physician, physic, technician or chemistry degree. However, these requirements are not constrained to the degrees within other subject science.

National resources for education and training in the country are mainly concentrated in the Academic Unit of Radioprotection (UARP), which is structurally dependent on the Institute of Public Health - Faculty of Medicine (FMED), University of the Republic (UdelaR). In 2018 a collaboration agreement was signed between the Ministry of Industry, Energy and Mining (MIEM) through the ARNR and the Faculty of Medicine (FMED), University of the Republic (UdelaR). This agreement establishes the objective of cooperation and collaboration for the action plan and implementation of the national training strategy in radiation protection, promoted by both institutions.

The course will be launched in the second half of 2020. It will be an ambition launch an OPR course in industrial practices in the second semester of 2021.

***Keywords**: radiation protection training, education.*

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PS3 (T3.8-0727)

Radiation Sources Fabricated from Tea Leaf and Educational Trial

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It is undeniable fact that everything on Earth contains more or less a number of naturally-occurring radioisotopes (natural radioisotopes), and emits radiation without exception. Typical examples are potassium chloride, chemical fertilizers, and kelp. These materials contain natural radioisotope potassium-40 (^{40}K). However, many people are unaware of their proximity to such natural radioisotopes.

In our previous studies, a compression and formation method was developed to solidify those materials into solid disk-shaped substances (solid disks) that emit radiation. Thus, the fabricated solid disks could be used in various educational radiation courses. Upon completing these courses, a number of students would accept the existence of natural radioisotopes. However, other students would have difficulty accepting this as they might feel those materials were uniquely special. Therefore, it is desired that materials that anyone are generally familiar with will be used as raw materials of the solid disks. Black teas are a potential material because they are commonly consumed all over the world and are considered to be a daily commodity for average people. So, more people may understand the existence of natural radioisotopes when they can measure the radiation emitted from solid disks fabricated from black teas by themselves.

In the present study, five brands of black teas were randomly purchased at supermarkets in Japan, and the compression and formation method was applied to them. Fabricated solid disks (tea solid disks) were examined in terms of weight, diameter, and thickness, and their gamma-ray spectra were also measured. Three dependence tests of radiation counts on time, distance, and shielding were then carried out with Geiger-Mueller (GM) survey meters to evaluate the tea solid disks as educational radiation sources. Furthermore, two trials of utilizing the tea solid disks were conducted. The first trial was to determine the existence of natural radioisotopes, and the other to comprehend the randomness of radiation counts to be represented by a Gaussian distribution curve.

As a result, all the tea solid disks fabricated in the present study were almost identical in dimension and weight regardless of individual fabrications and brands used as raw materials as well as kelps were, and contained natural potassium-40 (^{40}K) with 50 - 60 Bq/100 g. In regard to the three dependence measurement tests, the results obtained could only clarify the dependence on time, not distance or shielding. This is why radiation counts distinguishable from background radiation could be obtained when the tea solid disk was directly attached to the surface of the GM probe. The results of the two trials showed that the tea solid disks could be used as educational radiation sources.

Keywords: Radiation education, educational radiation source, black tea

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PS3 (T3.8-0909)

ENEN+ project - Attract, retain and develop new nuclear talents beyond academic curricula

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The main objectives of the ENEN+H2020 project are: attracting new talents to careers in the following nuclear disciplines: nuclear energy; medical physics; radiation protection, waste management and geological disposal. For the ENEN+ project it is imperative to provide activities focused on the three main target groups of potential talents: secondary school pupils, bachelor and master students and young professionals after graduation. Attractive basic information on careers in nuclear was developed and complemented with an EU wide competition of secondary school pupils. Several nuclear summer schools were organized for BSc students to attract them to pursue master education in nuclear which will be strengthened by increasing the level of academic preparation for bachelor students.

Career guidance with mobility support exceeding 1.000.000 EUR is envisioned. The ENEN+H2020 project gave financial support for more than 600 mobility actions during the three-year period. The mobility funding was accessible through competitive calls published on the ENEN website. More than 200 grants were given for nuclear engineering and safety, 90 for medical physicists, 50 for radiation protection and waste management. The 20% of the whole amount of money was spent on the organization of summer schools in different fields of nuclear sciences with more than 250 participants.

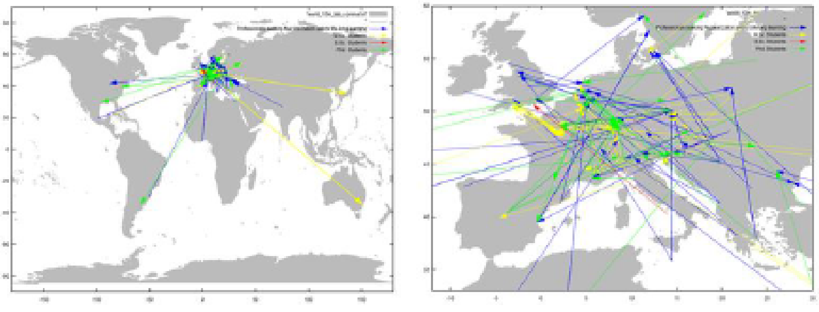

Fig. 1. ENEN+ activities in Europe (right side) and in the world (left side).

***Keywords**: nuclear sciences, young generation, grant program*

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PS3 (T3.8-1035)

7th International Conference on Education and Training in Radiological Protection (ETRAP): Groningen NL, March 2021

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It is widely recognized that education and training are essential pillars for applying ionizing radiation safely. Since 1999, a series of six conferences on Education and Training in Radiological Protection have been organized in Europe [1]. With the focus on benchmarking current practices and experiences on one hand, and on a harmonized approach towards education and training in Europe on the other, this series has proven to be very successful. The 7th ETRAP conference will be hosted by the Groningen Academy for Radiation Protection / University of Groningen, in close collaboration with SCK-CEN. The conference will be held from 23-26 March 2021 in the beautiful city of Groningen in The Netherlands. In this contribution we will update the congress participants on the details and program of the conference.

Keywords: *Education & Training, Conference*

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PS3 (3.A-0332)

Ethical core values and update of ICRP publication 62: Case example of brain imaging research

International Commission on Radiological Protection (ICRP) issued Publication 138 to identify four core ethical values: beneficence/non-maleficence, prudence, justice, and dignity; along with procedural values: accountability, transparency, and inclusiveness. This was achieved by continuous collaboration with the International Radiation Protection Association (IRPA) to provide international workshops and symposium. Then the ICRP set up a Task Group 109 to clarify "ethics in radiological protection for medical diagnosis and treatment" on the basis of ethical values identified in the Publication 138. However, "medical research ethics" is out of the scope of the terms of reference of this TG109, although "research" is now integral part of health care; and radiological protection system must have been developed on scientific evidence generated from research involving humans. Research ethics in the context of radiological protection was previously clarified in the Publication 62 in 1992 but this publication needs update reflecting not only recent progress of radiological science but also newly clarified ethical values as well as recent trend of health research ethics.

From this reason, to clarify some aspects which should be revisited in the Publication 62, this presentation shows one case example of brain imaging research involving healthy volunteers and patients with cognitive impairment but whose physical status is healthy, exceeding total effective dose 10 mSv, from which Publication 62 requires "social benefit" to be generated. We will make analysis focusing the following two points:

(1) How we should evaluate "social benefit" of this research of case example exceeding 10 mSv.

(2) Whether or not research result should be informed to study participant in view of "social benefit".

Finally, we will present overall discussion points considering the necessity of updating Publication 62, to be explored in collaborative partnership among IRPA, ICRP ant related communities. This would facilitate ethical conduct of research involving humans, in the field of radiological science, based on newly identified core and procedural ethical values underpinning radiological protection system.

PS3 (3.A-0662)

A Dilettante Takes a Second Look at ICRP Publication 138

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The ICRP issued Publication 138 entitled "Ethical foundations of the system of radiological protection" in 2018. This publication is the ICRP's attempt to address the ethical foundations of the system of radiological protection in an explicit manner. The intent among other goals, we to help "to clarify the inherent value judgements made in achieving the aim of the radiological protection system as underlined by the Commission in Publication 103". The publication addresses four core ethical values: beneficence/non-maleficence, prudence, justice, and dignity and links those values to the principles of radiological protection, namely justification, optimization, and limitation.

The author who by no means is an expert or even trained as an ethicist will present his thoughts upon reading the publication. He previously carried out such an analysis, so this is really his understanding and interpretation on a second, more comprehensive reading. The author will declare his worldview, which affects his interpretation of the publication, prior to discussing his thoughts.

Keywords: *ICRP 138, ethical foundations, nonexpert view*

ACKNOWLEDGMENTS

The presentation will only represent the views of the author.

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PS3 (3.A-1018)

Empathy as an additional value in the system of radiological protection

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Empathy can be defined as "the capability (or disposition) to immerse oneself in and to reflect upon the experiences, perspectives and contexts of others." It is often considered to be a character trait, or a personal quality – something that one either possesses, or not. But there is growing evidence that it can be consciously applied, and can be taught as well. It seems possible, therefore, to consider it a value, or an ethical principle which can be expected of professionals (Zölzer and Zölzer, 2020).

ICRP publication 138 addresses the "Ethical Foundation of the System of Radiological Protection". It identifies four core values that have been at the basis of the International Commission's recommendations over the decades. These are beneficence/non-maleficence, prudence, justice, and dignity. In addition, ICRP publication 138 suggests that the practical application of the system of radiological protection should be driven by three procedural values, accountability, transparency and inclusiveness (stakeholder participation). Empathy does not figure here, but it can be argued that it would be a valuable addition to the system – and also that it enjoys similar cross-cultural recognition as the other values, both core and procedural (Zölzer, 2016).

A review of the literature shows that empathy is called for in a broad spectrum of situations, ranging from the conversation of a doctor with a patient, to the assessment and management of public health problems, and way beyond (Zölzer and Zölzer, 2020). As concerns decision-making in radiological protection specifically, the idea promoted here of the place of empathy and the other procedural values mentioned above is as follows:

- The assessment of a radiological situation and the health problems accruing from it, must be driven mainly by *empathy*.
- For the communication about what has been ascertained, *transparency* is most crucial.
- In the consultation process about what should be done, *inclusiveness* (making sure that all stakeholders are involved) takes precedence.
- When decisions are taken and their effectiveness is evaluated, *accountability* is key.

In other words, it is suggested that people's concerns, their needs and wishes need to be taken seriously from the very beginning of any decision-making process, even if they are considered unfounded or exaggerated. Without empathy, our practice of beneficence and solidarity would be oddly limited.

Notabene: it is first and foremost the professional that is supposed to exhibit empathy towards the stakeholders. Instead, one of the Welcome Messages to the IRPA Congress seems to suggest "that the public (should) empathize with the issue of radiation risk and protection," which in my view is turning things upside down.

Keywords: ethics, core values, procedural values, accountability, transparency, stakeholder participation

ACKNOWLEDGMENTS

The author is indebted to the ICRP and the University of South Bohemia for financial support.

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PS3 (3.B-0590)

The Society for Radiological Protection (UK) Workstreams on Communicating Radiation Risk - Developing tools and guidance for the profession

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Across the Radiation Protection Profession and its allied fields, the communication of "Radiation Risk" is becoming an increasingly important skill. However achieving effective risk communication is becoming an increasingly challenging task given the somewhat negative public perception of radiation and conflicting views presented online and by both media and social media.

The Society for Radiological Protection (SRP) Annual Conference in May 2019 featured a workshop with a focus on "Communication of Radiation Risk in the Modern World". Three central scenarios were covered:

- Communicating to the Public Post a Nuclear / Radiological Incident
- Communicating Radiation Protection to Government / Local Authorities
- Communicating as part of Public Engagement Activities e.g. STEM

The workshop included technical talks and views from media specialists. The output of this workshop can be found in [1].

The workshop was well attended, and the feedback was promising with attendees requesting follow on workshops, and the development of subject specific guidance for Radiation Protection professionals.

IRPA is also currently showing an interest in this area and is developing a guidance document covering the guiding principles.

In summer 2019, SRP started a workstream aimed at developing a series of short specific user guides for the communication of radiation risk in certain scenarios, such as in support of Outreach, Emergency Preparedness or Medical Exposures.

The recent introduction of revised emergency planning legislation in the UK has the potential to result in the introduction of new or larger Detailed Emergency Planning Zones, and as such present an area of concern for the public, namely were they being protected adequately before, or is there a substantial increase in radiological risk? It was therefore decided that the first of the guides to be developed would be a "Guide to Communicating Radiation Risk in Emergency Preparedness".

The guide is to be developed via a workshop in November 2019 involving 15 attendees, including UK Government, Regulators, Media Specialists (including ex journalists, social media specialists, and specialists involved in communicating post actual incidents such at the Litvinenko poisoning), Nuclear Operators, UK Defense Operators, Radiation Transport Specialists and Local Authorities.

Following the workshop, SRP will pull together the draft "Guide to Communicating Radiation Risk in Emergency Preparedness" which will be sent to the wider participants for comment. Following resolution of any comments the guide will be formally published on the SRP website in PDF Format.

The proposed talk will provide an overview of the previous and ongoing work streams within SRP, and where the developed tools can be found for use by those working in the field of Radiation Protection.

PS3 (3.B-0590)

The Society for Radiological Protection (UK) Workstreams on Communicating Radiation Risk - Developing tools and guidance for the profession

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Keywords: Risk Communication[1], Emergency Preparednes[2], Public Engagement[3]

ACKNOWLEDGMENTS

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Poster Session T3

PS3 (3.B-1021)

Suggestions for Better Radiological Risk Communication with the Public using a New Index RAIN

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A new radiation scale is proposed. With empathy toward the vast majority of people who are not well versed in radiation and its possible harmful health effects, and thus suffering from misunderstanding that breeds unnecessary fear of radiation, the aim of proposing a new radiation scale, radiation index (RAIN), is to put the general public at ease with the magnitude of radiological risk.

RAIN is defined in dimensionless numbers that relate any specific radiation dose to a properly defined reference level. As RAIN is expressed in plain numbers without an attached scientific unit, the public will feel comfortable with its friendly look, which in turn should help them understand radiation dose levels easily and allay their anxieties about radiation exposure.

The correspondence between RAIN and the specific radiation dose is established. The equivalence will allow RAIN to serve as a common language of communication for the general public with which they can converse with radiological protection experts to discuss the level of radiological risk involved in various given radiation exposure situations. Such fruitful dialogues will ultimately enhance public acceptance of radiation hazards. The expanded awareness and proper understanding of radiological risk will empower the public to feel that they are not hopeless victims of radiation exposure.

Keywords: *Communication, Empathy, New radiation scale*

PS3 (3.B-1038)

Enhancing Stakeholder Participation in Radiological Protection – Findings and Recommendations of the European ENGAGE Project for Three Exposure Contexts

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Stakeholder engagement has become a key topic in the governance of radiological risks. This reflects the increased recognition and demand for stakeholder engagement in radiological protection decision-making, from both policy-makers and experts, as well as civil society organisations and potentially affected publics. Alongside legal requirements, generic guidelines and recommendations have been elaborated for stakeholder engagement in the context of specific exposure situations, recognizing the opportunities, pitfalls and potential solutions. The European project ENGAGE ("ENhancinG stAkeholder participation in the GovernancE of radiological risks for improved radiation protection and informed decision-making"), investigated how these demands and expectations are translated into participation practices at national and local levels. Based on this, it formulated recommendations for a more robust stakeholder engagement in radiological protection. ENGAGED compared and contrasted three contexts: medical exposure to ionising radiation, radiological emergency preparedness and response as well as exposure to indoor radon. The empirical focus in each of these contexts was threefold:

i) to analyse the *formal discourses prescribing or recommending engagement, as formulated in international and national legislation and guidelines and mobilised by different actors*, highlighting how stakeholders and stakeholder engagement are defined; what the underlying rationales are; and what is included or excluded from these frames;

ii) to highlight, through case studies and more systematic mapping exercises, the *forms of real or potential stakeholder engagement that can be observed in practice*, with attention to what the issues at stake are; how the outcomes and processes of participation are crafted; what are the main challenges and opportunities; and how these practices relate to the frames set by the legislative documents and guidelines analysed. The project took into account that invited participation by institutional actors is only one part of a more complex "ecology of participation", alongside citizen-led initiatives.

iii) to investigate through case studies *the role and potential benefit of radiological protection culture in facilitating stakeholder engagement and informed decision-making*, with identification of processes to build and transmit radiation protection culture, adapted to the specificities of different exposure situations.

This contribution summarises the recommendations discerned through the research and engagement activities carried out in the project [1]. These are relevant for radiological protection researchers, practitioners, policy makers, and civil society stakeholders.

Keywords: Stakeholder Engagement, Radiation Protection Culture, Radon, Medical, Emergency Preparedness and Response

ACKNOWLEDGMENTS

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PS3 (3.C-0670)

Challenges with a Non-Prescriptive Regulatory Control, South Africa

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Non prescriptive regulatory approach which can be attributed to the political, legal and industrial culture of the country, is a system in which safety goals are set, and it is the responsibility of the licensee to demonstrate compliance to the regulatory body by justifying any design code or specific acceptance criteria or safety measure that it uses. The broadness of the safety goals/requirements leads to the need for more content in the guidance documents and the need for adherence at the level of the guides, which are meant mainly to encourage common approach, rather than for enforcement.

For developing countries these safety goals are adopt, (not necessarily adapted) from the IAEA/ICRP recommendations, due to lack of resources to perform more research and customize these recommendations into befitting requirements. This is exacerbated by regulatory systems that rely on other bodies (governmental departments) for legalization (promulgation/ issuing) of the regulations, on which the guidance documents are based. Compared to the systems that are sufficiently equipped to promulgate/issue their own regulations (and guides in some cases) or on which the high level requirements are legislated, the non-prescriptive and the legally dependent regulatory system have no rigid credibility.

With expansion of the nuclear foot print worldwide and projects related to either power uprate or long term operation of old power plants, there is a growing need to be more clearer, inclusive and comprehensive in the regulations. Otherwise, the regulator relies on the expertise of the operator. This may result in a marginal regulatory control.

With the help of IRRS, INIR and other IAEA implements, the South African Regulator resolved to improve the level of detail of the regulations for safety and address the issue of the lack of adequate skills by using Technical Support Organisations (TSO's), with an intention of facilitating skills transfer to its staff. Therefore, moving towards being more prescriptive, in which the regulations are sufficiently detailed to provide the licensee with enough information on the general safety objectives.

When the South African National Nuclear Regulator (NNR) was established in 1999 through the National Nuclear Regulator Act, to provide for the protection of persons, environment and properties from the harmful effects arising from ionizing radiation, it was still faced with the challenge of becoming independent from the Atomic Energy Corporation, to which it was part. The NNR had the responsibility to set safety standards, determine authorization conditions and obtain assurance of compliance. At that stage, the NNR adopted a non-prescriptive approach: the applicant/authorization holder was expected to submit a safety case demonstrating safety against the safety goals. However, the lessons learnt from the operation of nuclear facilities, the Pebble Bed Moderator Reactor project, and the conduct of a comprehensive self-assessment using IAEA safety standards led the Regulator to modify its approach. New draft regulations more detailed were developed as well as a comprehensive suite of regulatory guidance documents for support.

Keywords: Regulations, non-prescriptive, prescriptive

PS3 (3.C-0698)

Methodology for the Development of Competencies of Regulators of Medical and Industrial Radiological Applications

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Conscious of the strategic importance of strengthening the competencies of the regulatory staff that exercises their functions in medical and industrial applications, the Ibero-American Forum of Radiological and Nuclear Regulatory Agencies (FORO) carried out the project "Competences of the Staff of Regulatory Bodies in Medical and Industrial Radiological Applications", which was an opportunity for the member countries of FORO to compare all the elements that constitute a system of training, acquisition and management of competencies (knowledge, skills, attitudes) in radiation protection.

Some specific aspects considered in the technical exchange were the following:

- Mechanisms for the development, implementation and management of competencies
- Design of competency matrices
- Mechanisms of recruitment and training of technical staff
- Existing resources (own, external, national, regional and international)
- Regional cooperation
- Use of international support (IAEA and others)
- Participation in international programmes and projects (IAEA and others)

The objective of this paper is to present a methodology to determine the competencies required by the staff of the regulatory body according to its functions in medical and industrial radiological applications. The applicability of this methodology will vary from country to country depending on the nature and characteristics of its facilities and activities in the medical and industrial areas and the degree achieved in the implementation and development of national strategies for the acquisition and management of regulatory competencies.

It is expected that the application of this methodology will contribute significantly to the development and growth of regulatory infrastructures, through the assurance of the adequate profiles, competencies and training of its staff in order to guarantee its effectiveness and efficiency in the performance of its duties.

This work has been funded by the Ibero American Forum of Nuclear and Radiation Safety of Regulatory Agencies (the FORO) and carried out under its technical programme.

KEYWORDS: Competencies; Regulatory Bodies; medical and industrial applications

PS3 (3.D-0845)

Activities for Fostering Safety Culture in J-PARC

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The Japan Proton Accelerator Research Complex (J-PARC) is a multi-purpose research facility with using MW-class proton beams to produce various secondary particles. Radioactive material leak incident occurred at J-PARC in 2013 [1], which caused radiation exposure of workers and radioactivity release outside controlled area. After this incident, the J-PARC Center struggled for improving mindset for safety with all staff members. Activities for fostering radiation safety culture at J-PARC are introduced.

Various safety trainings for staff members, contactors, and experimental users were strengthened after the incident in 2013. In terms of radiation safety, several kinds of e-learning trainings had been carried out as voluntary safety trainings in addition to statutory trainings for all rad-workers. The themes of these e-learning courses were "Basics on radiation and radioactivity" (2015), "Handling rad-contaminated materials" (2016), "Using GM survey meter for checking surface contamination" (2017), and "Body contamination and recovery actions" (2018). Reporting procedures in emergency was also learned as the topic in another e-learning courses. Figure 1 shows an example of the slides dealing with the situation for reporting when a radiation accident occurs at J-PARC facilities. Furthermore, several kinds of practical trainings were also carried out to improve technical skills for conducting safe rad-works.

Fig. 1. E-learning contents on reporting procedures in emergency.

All staff members and experimental users in J-PARC are requested to carry the J-PARC Safety Card which describes the slogan "Science with Safety" along with reporting procedure in emergency and the site map for emergency assembly areas. Further, "Mindful of others" campaign, which encourage paying attention and speaking to others when workers find an act of danger, has been continuously promoted since 2016. Another topical event concerning safety culture is "the J-PARC Safety Day" set around May 23 on which the radioactive material leak incident occurred in 2013. The Safety Information Exchange Meeting and the Workshop for Fostering Safety Culture are held on the day. The main speakers for the workshops are invited from various field, for example, Japan Railway Company, Tokyo Disney Resort, Toyota Motor, and medical field. We believe that their talks introducing unique approaches for safety aid J-PARC members to recognize the importance of safety and to consider their own standpoint and approaches aiming for safety improvements.

Symposium on Safety in Accelerator Facilities [2] is held every year to share the information and experiences on safety issues with 100-150 participants attending from accelerator facilities in Japan and overseas. The main topics regarding radiation safety in the symposia are management of induced radioactivity (2016), emergency response (2017), radiation safety education (2018), personal dose management (2019), and interlock system at accelerator facility (2020). Reports on lessons learned from accidents and troubles are important topics welcomed to be shared in the symposium.

Keywords: Safety culture, Safety training, Sharing information

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PS3 (3.D-0851)

Motivation and Activities on Radiation Safety Culture Improvement in Utokyo Leaded by Radiation Safety Promotion Manager

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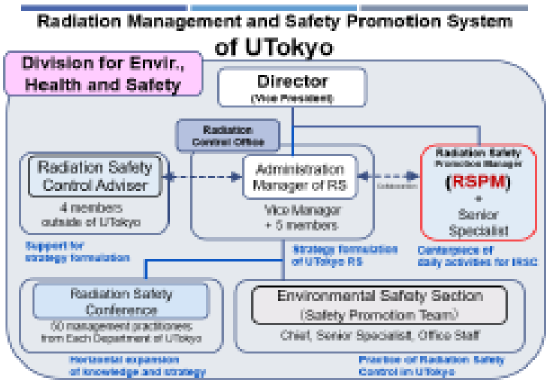
International organizations such as International Atomic Energy Agency (IAEA) and International Radiation Protection Association (IRPA) etc, continue their deep discussion on the improvement of radiation safety culture (IRSC) in various types of radiation related facilities. In Japan, the regulatory body on radiation application has also been considering the way to make their effective guidance and inspection to the facilities based on the revised law enrolled in April of 2018.



In parallel with these movements and background, the president of The University of Tokyo (UTokyo) decided to develop and started the new system for radiation safety management from 1st July of 2017. UTokyo owns more than 3,000 radiation workers and operates 30 licensees on radiation and nuclear applications. A symbolic key is Radiation Safety Promotion Manager (RSPM), which is the UTokyo original, enrolled in Division for Environment, Health and Safety. The main roles of RSPM expected are, [Mission I] to be the centerpiece of daily activities for IRSC in UTokyo, and to be the center point of aggregating and exchanging information and opinion between UTokyo and (1) regulatory body on radiation and nuclear application, (2) police and fire stations around UTokyo, (3) local governments around UTokyo, (4) other universities, institutes and organizations, etc., and [Mission II] to lead UTokyo to be a top runner in domestic and international activities in the field of radiation safety management.

This presentation shows the challenging activities of RSPM in 2017-2019.

Table 1. Main Examples of Activity Scope of RSPM and Senior Specialist of UTokyo

Mission I	Mission II
✓ Drafting UTokyo radiation management policy	✓ Participating activities of domestic and foreign organizations as a radiation protection expert
✓ Support for safety meetings and activities of university-wide	✓ Collaboration with other organizations as an academic member
✓ Information collection on the management status of each facility	✓ Expansion activities on radiation safety culture development activities
✓ Advice and guidance on management policy formulation for each facility	✓ Activities related to human resource development in the field of radiation protection and safety management
✓ Hosting of workshop of nuclear fuel material safety management, workshop for managers of X-ray irradiators, radiation safety promotion lecture of university-wide, etc	
✓ Planning and publishing a textbook for radiation workers on annual retraining	
✓ Off-campus contact as a radiation safety specialist	
✓ Guidance and support for accidents and troubles in UTokyo	

Keywords: Radiation Safety Culture, Radiation Management, Leadership

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PS3 (3.D-0939)

Safety Culture: Lessons Learnt form Historical Accidents

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Over the last decades, serious nuclear accidents occurred from which nuclear organizations can draw lessons learnt and apply them in the future for a safer use of nuclear technology. In this presentation, the Chernobyl and the JCO accidents were again studied, and lessons learnt for improvement of safety culture would be drawn through new viewpoints and approaches after the Fukushima Daiich accident.

The responsibility of the Chernobyl accident did not fall only on the operators but also on the related organizations, Gosatomnadzor, the State Sanitary Inspection, and the Council of Ministers. Chernobyl reactors, RBMK-100, were originally designed by the Ministry of Machine Building who did not provide details on the reactor's instability and danger during low-power operation to the Ministry of Atomic Energy. This factor in addition to the operator insistence on pursuing the safety test under conditions that caused instability of the reactor were considered as the immediate causes of the accident. The accident triggered a lot of attention to Safety Culture at that time, such as publications of International Organizations (e.g. IAEA) to stress on the importance of Safety Culture.

In spite of the Safety Culture recognition at that time, the JCO criticality accident in Japan occurred only few years after the Chernobyl accident. JCO was a fuel fabrication company which main task was to convert Uranium compounds and to produce low enriched LWR's fuels. JCO was also engaged in high enriched uranium nitrate conversion for the fast reactor. The Science and Technology Agency (STA) and the Atomic Safety Commissions (ASC) of Japan were responsible of regulating the operations of JCO, by taking into consideration several measures including the geometric safety vessels, the concentration limitation and the mass limits. The JCO accident occurred due to violation of licensed processes by JCO staffs that used an unauthorized process by pouring excess amount of uranium nitrate solution into the sedimentation tank that was not geometric safe design. They say that these were the direct causes of the criticality accident as well as other factors as lack of education and training in criticality and its causes. IAEA also pointed out almost the same direct causes and background of the accidents in the official report as the official report of Japan. Nevertheless, as the result, it seems main responsibility felt only on the performance of the safety senior chief director.

The level of Safety Culture in organizations is highly important for managing organizations and ensuring safety of the operation. Based on lesson learnt by comparing the two accidents, it should be noted and strengthened that strong commitment and dedication of high decision-making bodies on safety would be one of the significant factors that could ensure a high level of safety culture preventing that severe accidents happen in the future.

Keywords: Safety culture, Nuclear accidents

PS3 (3.D-0977)

Radiation Safety Culture in the UK Medical Sector: An update on a top to bottom strategy

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In 2011 UK professional bodies established a number of sectorial working parties to provide guidance on the improvement of radiation safety (RS) culture in the workplace. The medical sector provides unique challenges in this regard, and the remit of the medical group was to review the current state of RS culture and to develop a framework for improvement. The review of RS culture was based on measurable indicators, including data from regulatory inspections, personal monitoring data and incident data. An online survey to capture the RS-related views and experience of hospital staff at all levels was carried out, and the responses provided a wealth of information on RS awareness and implementation across the country. The framework for improving RS culture included both 'top-down' initiatives to engage management and regulators, and 'bottom-up' initiatives relating to engagement and training of different staff groups. A 'Ten-point Assessment' on what constitutes a good approach to medical radiation safety culture was published, which provides a tool for management to assess RS culture in the work place and has potential use in regulatory inspections in the UK. Now ten years on this poster reflects on progress made.

Poster Session T3

PS3 (3.D-1037)

Work of the IRPA Task Group on Radiation Protection Safety Culture in the Higher Education, Research and Teaching (HERT) Sectors

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The Task Group (TG) was established in 2017 with the objective of encouraging the awareness and development of a strong radiation protection safety culture in the Higher Education, Research and Teaching (HERT) Sector throughout the World. Currently, it has 10 members from various IRPA Associate Societies, and the link to the IRPA EC is maintained by Bernard Le Guen.
The TG work mainly via e-mail but held a meeting at the 5th European IRPA Congress in The Hague in May 2018 to review progress and develop an action plan moving forward. This was attended by 3 members of the TG plus the President of IRPA.
The TG has produced a document containing '10 points' for developing a good Radiation Safety Culture in HERT sectors. Steps have been taken to translate this document from English into other languages notably at this stage Spanish and Japanese for dissemination in Latin America and Japan. The '10 points' were presented at a meeting of The Ghana Atomic Energy Commission Radiation Protection Institute and were received well.
In Japan, survey work has been conducted on "Trial Activities on RS Culture Improvement in the Japanese HERT Sector". Results from some organizations relating to this work will be shared at the IRPA15 Congress
In the UK, the Association of University Radiation Protection Officers (AURPO) developed and conducted a survey to assess the current state of RS culture in UK HERT sectors. The preliminary results of this survey were presented at the 5th European IRPA Congress in The Hague in May 2018. A paper for the Journal of Radiation Protection is in development.
The TG aspires to establish a dedicated HERT RS Culture webpage on IRPA website with a view to populating this page with useful resources that can be downloaded and adopted by IRPA Associate Societies throughout the world – such as: the '10 points' document, survey questionnaires developed in UK and Japan, and a number of journal publications related to HERT RS culture development.
The TG also plans to develop a 'Radiation Safety Culture Toolkit' for the HERT sectors which could be downloaded from the webpage by IRPA AS and other organisations for use in their respective countries.

***Keywords**: Research, Safety, Culture*

PS3 (3.D-1136)

Radiation Safety Culture – Nuclear Sector

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The Society for Radiological Protection's Nuclear Industries Committee undertook a review in 2019 of the current practices across the UK nuclear industry which contribute to radiation safety culture. Drawing from a variety of nuclear sites, these have been collated into a baseline showing the benefits and limitations of each. The review covered industry wide activities that directly contribute to radiation safety culture as well as site specific activities. It also considered the future challenges to radiation safety culture. This work will be used to guide future SRP work on radiation safety culture and will form part of SRP's input to IRPA's work on this topic.

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Keywords: *Culture, Nuclear*

PS4 (T4.1-0312)

Optimization of ^{67}Cu production via $^{70}Zn(p,\alpha)$ reaction using computer simulation

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Radioisotopes emitting short-range highly ionizing radiation such as β-particles are of increasing significance in in-situ radiotherapy. Among β-particle emitting radioisotopes, ^{67}Cu is an attractive radioisotope for cancer treatment. However, clinical application of ^{67}Cu is not yet common due to the difficulty in its availability with sufficient amounts. One of general methods to produce ^{67}Cu is via $^{70}Zn(p,\alpha)^{67}Cu$ nuclear reaction using a cyclotron. We performed Monte Carlo simulations using MCNPX code for the optimization of ^{67}Cu production with a cyclotron. A multi-physics simulation software, COMSOL Multiphysics®, was also used to estimate the heat transfer in the target and to confirm that the zinc target is able to withstand the high-current proton beam irradiation, which needs to be kept below the melting point of the zinc, thereby, to prevent possible loss of the expensive target material. The maximum production yield of ^{67}Cu for a 30 MeV proton beam was calculated to be 4.547 MBq/μAh for the zinc target mass of 1.5 g as shown in table 1. It was also confirmed that the maximum temperature at the zinc layer in all the cases under the beam current of 200 μA was kept below its melting point. The result was compared with another particle transport code, SRIM, and an experimental results from the previous study. It is expected that, with an optimal beam condition, a stable production of ^{67}Cu via $^{70}Zn(p,\alpha)^{67}Cu$ reaction using a 30 MeV medical cyclotron is possible, with quantities sufficient for therapeutic applications including animal study and clinical trial.

Table 1. Comparison of production yield of ^{67}Cu between calculated results from this work and experimental results at EOB

Zinc mass(mg)	Incident energy (MeV)	Energy range (MeV)	Production yield (MBq/μAh)		
			MCNPX	SRIM	Experimental yield
500	30	30 → 24.12	1.582	1.13	
	28	28 → 21.73	1.883	1.35	
	23	23 → 15.37	2.641	2.53	
	18	18 → 7.68	2.374	2.60	
1000	30	30 → 16.86	3.468	3.29	
	28	28 → 13.61	3.727	4.26	
	23	23 → 0.00	3.680	4.09	
	18	18 → 0.00	2.465	2.63	
1500	30	30 → 5.34	4.547	5.48	4.87 (Kastleiner et al., 1999)
	28	28 → 0.00	4.502	5.13	
	23	23 → 0.00	3.779	4.09	
	18	18 → 0.00	2.466	2.63	2.35 (Kastleiner et al., 1999)

Keywords: *^{67}Cu, Simulation, Cyclotron*

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PS4 (T4.1-0442)

Optimization of image quality and radiation dose minimization in bone X-ray radiography

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In this study, the goal was to optimize image quality and minimize radiation dose in bone X-ray radiography procedure. Investigations were conducted to determine optimum quality images of the dedicated phantoms and samples at minimized initial radiation dose. In this regard, the main problem in this study was to find X-ray scanning parameters (voltage, current, exposure time and focal-sample distance (FSD)) that give high quality images with minimal initial radiation dose. Micro-focus X-ray machine, identifinder ultra and ImageJ software were used for acquisition of images, radiation dose monitoring and evaluation of image data sets, respectively. For the optimization process, the weight sum technique was applied in the analysis of the acquired radiographs. The best compromise between the optimal image quality and minimized radiation dose was determined. The most optimal image quality and minimized radiation dose occurred with X-ray tube voltage of 150 kV, X-ray current of 100μA, X-ray exposure time of 1 sec and FSD of 65 cm.

PS4 (T4.1-0758)

Design and preoperational radiological protection in a proton therapy center

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Introduction: The facility consists of a proton beam therapy (PBT) system model "Expandable One Gantry System" (Hitachi), with the ability to accelerate protons with energies from 70 to 230 MeV. The main areas are: accelerator room (AR), gantry room (GR), treatment room (TR), treatment control room (TCR), and accelerator control room (ACR). The facility was designed to optimize the staff and public radiation safety. The objectives are to present the facility design and to evaluate the preoperational radiological protection results during the commissioning period of the PBT system.

Methods: Safety system includes (1) a set of area crash buttons and area search buttons, (2) personal keys to enter AR and GR, (3) AR door interlocks related to a time delay after the beam stops and to the radiation levels inside the AR, and (4) panels showing "area ready for beam" and beam status.
The facility has neutron, gamma and contamination portable monitors, and a gamma spectrometer for radionuclide identification due to neutron activation of the materials. A well was also built to retain possible water leaks from the cooling system. Air and water activation are measured in the ventilation system and inside the well by means of gamma detectors.
A radiation area monitoring system records and displays the neutron and gamma dose-equivalent rates (H*(10)) in 4 locations of the facility (AR, GR, TR and ACR), measured with Wendy-2 and FHT 612-10 detectors. Operational area radiation dose is also monitored around the facility, in 93 points, with passive detectors calibrated in Hp(10) (gamma thermoluminescent (TLD) and neutron track detectors).
Dose assessment of radiation-exposed staff is performed with gamma TLD, and neutron track operational dosimetry. When entering into the AR and GR, staff must also use electronic personal dosimeter (EPD). Personnel working with system-activated parts must use ring dosimeters (Hp(0.07)).

Results: During the commissioning period (4 months) of the PBT system, no air or water activation were detected. Maximum H*(10) rates measured during PBT system commissioning are shown in table 1. Area monitoring, with passive dosimeters, corresponded to background levels, except at 10 points of the facility where a maximum dose of 0.18 mSv/month was registered.
Staff's Hp(0.07), and gamma and neutron Hp(10) values corresponded to background, and maximum accumulated and EPD doses were 0.80 µSv.

Table 1. Maximum H*(10) (µSv/h) rates measured

	AR	GR	TR	TCR
Neutron	2964	1633	2232	0.38
Gamma	159	41	81	0.38

Conclusion: The preoperational results show that the proton therapy facility has been designed to meet dose limits and ensure operational and public radiation protection.

Keywords: *Proton therapy, facility, neutron*

PS4 (T4.1-0945)

Influence of Data Composition in Deep Learning Automatic Tumor Segmentation System

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Gross tumor volume (GTV) segmentation is a significant factor to decide the irradiation dose in the radiotherapy process. Some previous researches have performed the deep learning (DL) based GTV segmentation to automize the segmentation process. In this study, we performed the DL network training with different GTV conditions to acquire the non-biased automatic GTV segmentation process.

Computed tomography (CT) and positron emission tomography (PET) data were obtained from the cancer imaging archive (TCIA). Each PET and CT were converted to 1000 slices 2D grayscale image pair and GTV ground truth label was acquired from TCIA treatment planning files. DL network was constructed in the form of U-shape network. 2D convolution layers with 3×3 and 1×1 kernel and 4 pooling-unpooling steps were adapted to the network. After 4 pooling process, input data is shrunk up to 1/256 of original area and channel is expanded up to 1024. Training process was performed twice with 2 different data setup: Training 1 was performed with 2D GTV paired slices, and training 2 was performed from GTV and non-GTV slices. All dataset was pre-processed with same methods and augmented using image flip and rotation. Accuracy of each training result was validated using dice similarity coefficient (DSC).

As the result of training 1 and 2, average DSC of each DL networks were shown the similar value. However, in training curve comparison, training 2 was shown the better tolerance at data overfitting issues. It means that DL training with non-tumor included dataset can improve the GTV segmentation accuracy than only tumor data based training. To obtain more accurate results, additional data acquisition and network optimization are required as the future study.

***Keywords**: Tumor segmentation, Deep learning, Data processing*

PS4 (T4.2-0058)

Estimation of the eye lens dose in interventional radiology

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Since February 2018, the annual equivalent eye lens dose limit has been decreased from 150 mSv to 20 mSv in The Netherlands. This is especially relevant for interventional radiology, a medical specialization involving minimal invasive image-guided procedures to diagnose and treat diseases. The annual dose can add up to a substantial individual staff member's work-related radiation burden. Although wearing lead glasses yields a reduction factor of approximately 3, the efficacy depends mainly on the position of the operator relative to the source (patient) and the type of procedure [1]. Therefore, the purpose of this study was to evaluate the eye lens dose of interventional radiologists and compare it to personal dosimeter (PDM) readouts.

The measurement setup consisted of cumulative and procedure-specific measurements using 11 eye lens dose meters of Mirion Dosimetry Services (Arnhem, The Netherlands). For a period of four weeks four interventionist wore an eye lens dose meter during all procedures yielding the cumulative eye lens dose. The cumulative eye lens dose was compared to the cumulative PDM readouts. Additionally, the eye lens dose of seven individual procedures (3 FEVAR, 2 SIRT and 2 TIPS) were measured separately and compared to an estimated equivalent dose based on dose-area-product (DAP) and position of the interventionalist taking distance and shielding factors into account [2].

The cumulative eye lens dose was lower (-83%, mean) than the cumulative PDM result for all four interventionalists (Table 1). The eye lens dose of the TIPS procedures (1.20 mSv and 0.60 mSv) was higher than the estimated equivalent dose (0.13 mSv and 0.06 mSv). Also for the FEVAR (0.09 mSv vs 0.05 mSv) and SIRT (0.08 mSv vs 0.01 mSv) procedures the eye lens dose was higher than the estimated equivalent dose. The results of three eye lens dosimeters (2 FEVAR and 1 SIRT procedure) were below the measurement threshold of the dosimeter (<0.01 mGy) and therefore excluded from analysis.

In conclusion, the personal dosimeter overestimates the eye lens dose in daily clinical interventional radiology by on average 83%.

Table 1. Comparison of eye lens dose with PDM during 4 weeks period

Measurement	Cumulative Eye lens dose (mSv)	Cumulative Personal Dosimeter (mGy)
Interventionalist 1	0.40	0.86
Interventionalist 2	1.69	3.29
Interventionalist 3	0.22	0.50
Interventionalist 4	3.19	5.40

***Keywords**: eye lens dose, interventional radiology, dose-area product*

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PS4 (T4.2-0068)

Study of workstations and Radiation Exposure Assessment of workers at the Nuclear Medicine Department of the CHU-Andohatapenaka-Antananarivo

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Nuclear medicine involves handling of radioactive materials (radiopharmaceuticals, radioactive calibration sources...) that can give rise to external and/or internal exposure of workers. Therefore, it is important to evaluate the individual exposure which is one of the main elements of radiation protection of workers exposed to ionizing radiation and which aims to assess the doses received at the whole organism and to implement the principle of optimization that exposures should be kept as low as reasonably achievable.

The present study focuses on dose measurement and radiation exposure assessment in all workstations within the Scintigraphy in the Nuclear Medicine Department of the University Hospital Center of Andohatapenaka, Antananarivo, Madagascar. Actually, no effective method of individual monitoring is available at this Department, thus a method based on workplace monitoring has been developed for an assessment of exposure. This enables to assess the workers exposition to radiation.

Measurements were made in each workstation using the RAM R-200 dose rate meter of the Laboratory of Nuclear Physics and Physics of the Environment (LPNPE) – University of Antananarivo. According to the workstation measurement results, it was found that the dose evaluated in this nuclear medicine department is lower than the occupational exposure limit recommended by the International Commission on Radiological Protection (ICRP). Indeed, the dose evaluated in this service is less than 20 mSv / year for the worker.

Keywords: Nuclear medicine, radiation exposure assessment

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PS4 (T4.2-0139)

Discussion in regard to radiation dose assessment and radiation protect measures for surgeons and staff during Balloon Kyphoplasty in orthopedic

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In the field of orthopedic, fluoroscopy is essential mainly when performing bone joint procedures. When fluoroscopy is used, surgeons as well as patients are at risk of X-ray exposure. The trunk part is protected by wearing lead protectors, but there are currently no measures for reducing radiation exposure of the surgeon's fingers and eye lens. Balloon kyphoplasty is a short-time procedure (within approximately one hour) under consecutive fluoroscopy, and the exposure dose of the surgeon's finger skin and eye lens is expected to be high compared to other procedures performed using pulse fluoroscopy mode. In addition, since the number of surgeons that can perform Balloon Kyphoplasty is limited, it is considered to be one of the procedures causing an increase in the cumulative dose of the surgeons. In this study, the exposure dose during procedures was measured using a ring dosimeter for fingers and a DOSIRIS dosimeter for eye lens. By conducting a prospective study for about six months in a joint study between National Hospital Organization Disaster Medical Center and Tokyo Health Care University, we will clarify the actual situation regarding the equivalent doses of the surgeon's finger skin and eye lens, and consider the future challenges. We will also discuss radiation protection measures by measuring radiation doses of surgical assistants exposed scattered X-rays and clarifying changes in consciousness regarding occupational exposures before and after dose measurements.

Keywords: *dosimeter for eye lens, dosimeter for fingers, Balloon Kyphoplasty*

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PS4 (T4.2-0142)

Radiation protection in cardiology with StarTable and Radpad

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The interventional cardiologist receives one of the largest exposures to scattering radiation. Without correction for wearing the lead apron, this can lead to an effective dose of more than 20 mSv/year. Different measures can be applied to reduce the exposure. In Noordwest the Radpad (Simons and Orrison, 2004) and Startable (Lunt, 2018) are used to shield the interventional cardiologist from radiation scattered by the patient.

The department of interventional cardiology has purchased Startable to have a proper position of the arm for punction the artery. One of the other advantages of Startable is the inherent shielding of the device placed between the patient and the cardiologist. The RadPad is obtained for its shielding.

The radiation protection department has questioned the amount of shielding of the StarTable and RadPad. Since there was little literature about the shielding of both devices, we decided to perform measurements. The measurements are carried out in four different settings for both position of the interventional cardiologist and the assistant. The four settings are: without both devices, with StarTable, with RadPad and with both devices. The eight most important tube positions in interventional cardiology are taken in consideration.

Another question that is answered in this investigation is whether the dose of the patient is raised, when using the RadPad. The RadPad is theoretically scattering the radiation coming out of the patient. Hence measurements are carried between the patient and the RadPad.

The results of the measurements are given in the presentation.

Keywords: Cardiology, Radpad, Starboard

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PS4 (T4.2-0147)

Box and whiskers method to assess the risk of the institutional dosimetry

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Introduction: Monitoring the exposure of ionizing radiation at the institutional level is as important as the personal dosimetry of the workers. Currently, dosimetry focuses on the worker through records between annual, quarterly or monthly periods, but at the institutional level is not analyzed. To guarantee a radiological protection program that is sustained over time, it is necessary to know in which area of the institution a greater resource of the estimated ones is required due to the risk that the practice merits. Therefore, for an optimization of the practice and the radiological exposure, it is necessary to characterize the areas and people in a dosimetry magnitude that is comparable between areas and periods. **Method:** It proposes a new method to show the global data in graph of box and whiskers where all the data of a worker of one year or more are synthesized. This allows to knows the dosimetry distribution and its width means correct level of dosimetry as occupational exposure practice and should be accompanied with the median. **Results:** At the Hospital Universitario del Valle 38 occupational radiation workers were studied for last 5 years using the method of graph of box and whiskers with monitoring monthly (N=2280 data of registers of dosimetry).

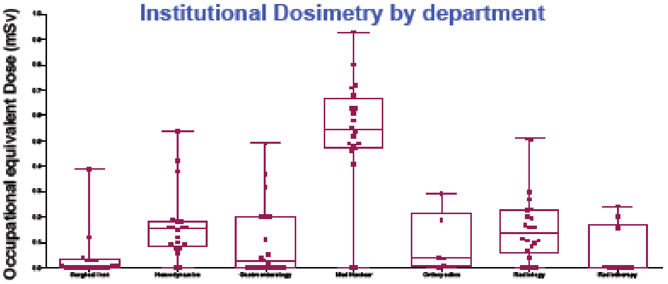

Fig 1: Institutional dosimetry distribution by department in 2018. n= 456 registers of dosimetry with 38 occupational workers follow monthly.

The Fig 1 shows the distribution of doses in each department of hospital, it permits analyze visually in which department resources should be directed together support with the tracking of practice inside each department of hospital. It could be determined with changes in the mean from each worker. When the mean of one worker is a little high and the Dosimetry Range are width, they must receive a retrain about your practice.

Conclusion: The graph of box and whiskers permits easer analyze visually where to focus resources and quick decision making and optima practice personal o institutional dosimetry the width of Dosimetry Range must not been compared with the occupational limit, must be much small what this value. The analysis among workers of the same department and the dosimetry between the different departments allows to evaluate the culture of radiological safety and evaluate the program of institutional radiological protection.

Keywords: Institutional dosimetry, *personal dosimetry,* box and whiskers

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PS4 (T4.2-0470)

Highlighting the use of Personal Radiation Dosimeters correctly in a clinical setting

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Under the Ionizing Radiation Regulations 2017 (IRR17) and Health and Safety at Work Act (HSWA) it is a legal requirement that all personal radiation dosimeters issued to any persons working with ionizing radiation must be worn when working with ionizing radiation. Personal radiation dosimeters allow staff radiation doses to be monitored and these results can be used further manage staff radiation safety accordingly. The position in which the personal radiation dosimeters are positioned on the body and with respect to any Personal Protective Equipment (PPE) such as lead aprons must be as stated by the Approved Dosimtery Service (ADS) from which the dosimeters are issued. Incorrect positioning of the personal radiation monitors could lead to inaccurate personal dose readings and thus impact on radiation safety. To ensure that personal radiation monitors are worn correctly effective instruction and training is key. Two posters were created, one to highlight the correct position to wear the dosimeters (Fig. 1.) and one to highlight the key points to further promote good practice. The posters have been placed outside rooms such as interventional radiation suites, next to the location of the personal radiation dosimeter storage and in a theatre setting next to the lead aprons/PPE. The results of the personal radiation dosimeters will be reviewed over a three month period and compared to a three month period prior to the placement of the posters. These results will be analyzed and presented in the results section of the project as one of the means to measure the effectiveness of the posters. Staff will also be asked to feedback via a questionnaire as to how useful they found the posters. To ensure radiation safety of staff promoting the correct use of the tools to measure dose and protect staff is vital.

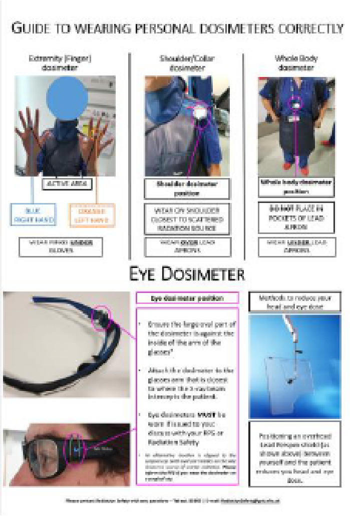

Fig. 1. Guide to wearing personal radiation dosimeters correctly

Keywords*: personal radiation dosimeter, staff radiation safety, IRR 17*

PS4 (T4.2-0573)

Implementing Radiation Safety Practices During the Decline, Expiration and Post Mortem of an I-131 Thyroid Ablation Patient

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250 mCi of I-131 was administered to a patient for thyroid ablation radiotherapy under the direction of an authorized user physician on an inpatient basis utilizing accepted radiation safety practices. At the time of administration, the patient's dose rate was 70 mR/hr at one meter.

Radiation Safety was notified approximately 12 hours post administration that the patient had vomited and was in respiratory distress. At that time, the patient's exposure rate was 30 mR/hr at one meter. Computed Tomography imaging and transfer to ICU was required. Radiation Safety staff members prepared a room in ICU with standard universal precautions and all appropriate staff members were educated. When the patient arrived at ICU they were intubated, catheterized, and had a PICC line placed. The patient's health continued to decline. 24 hours post administration, the patient's exposure rate was 30 mR/hr at one meter.

Radiation Safety maintained radiation control for family members and all healthcare personnel who interacted with the patient by setting up clean and contaminated boundaries for the patient room. Everything leaving the patient room (healthcare employees, family members, equipment and lab samples) was surveyed. The patient was catheterized, and the collection bag was surveyed indicating minimal kidney function from clearing iodine. Radiation safety was involved with the unsuccessful attempt at dialysis of the patient. The patient continued to decline. Radiation safety worked with the family to develop a time and distance strategy to maintain regulatory requirements while allowing family members time for their final goodbyes.

Upon the patient's passing, the body was cleared of all tubing, leads, IV's, clothing and then washed by nursing and radiation staff prior to being double bagged (the first bag had become contaminated therefore a second body bag was necessary). The patient measured 30 mR/hr at one meter. Transplant of the patient's corneas was denied. The patient's body was tagged and posted with a CRAM sign and then transferred to the hospital morgue and subsequently, the funeral home. The patient's room was thoroughly decontaminated. The hospital security and morgue staff as well as the funeral home staff were provided radiation safety education.

Radiation safety performed a site visit to the funeral home following the embalming process to ensure safety for funeral services. Radiation Safety verified that all work surfaces and tools were not contaminated, and all exposure readings measured background. Biohazard waste from the embalming process was held in long term storage. Wipes and surveys were performed on the patient in order to determine if an open casket could be allowed. Final readings at the funeral home measured less than 2 mR/hr. It was determined that an open casket was possible for family members, with a closed casket and regular burial for the funeral service.

Keywords: *Radiation Safety, Iodine 131, Patient Death*

PS4 (T4.2-0599)

Radiological safety management in a nuclear medicine department

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In our country, it is established that regulatory authorizations for the safe realization of practices, that imply the exposure to ionizing radiation of occupationally exposed personnel, patients and public, will base the safety of the realization of referred practices with solid technological criteria, backed for reliable management and organization characteristics in the use of the sources attached to the practices.

This presentation shows, how is materialized, in a nuclear medicine department, that performs "in vitro" diagnosis, gammagrafic studies and metabolic therapy, both, ambulatory and with hospitalization, using radioactive substances, the consideration as a determining criterion, the radiation protection of the occupationally exposed personnel, public and patients in the entire process, beginning from the previous identification of yearly demand of radioactive material, both the activity and the frequency of reception, through the measures for the safe transportation of radioactive material, carried out on its own hospital, storage and internal dispensing of radiopharmaceuticals, the administration and administered patients flow within the department, as well as the collection, classification, storage and release of generated liquid and solid radioactive wastes.

It also expresses the way which quality control is carried out on the operating equipment and the safety important elements that affect the optimization of the exposures of staff, patients and public, as well as the system and the results of the individual dosimetric surveillance and area monitoring of job and surrounding places with public access.

Finally, the main elements of the Safety Review of the practice are presented, which has supported the ownership of the Institutional Operation License, currently in force, for department, which provides important healthy services to a region with a high demand, far away from institutions that provide radioactive material supply and radiation safety support services.

Keywords: *radiation protection management*

PS4 (T4.2-0658)

Use of the "risk matrix" method in the safety assessment of radiotherapy treatments in Cuba

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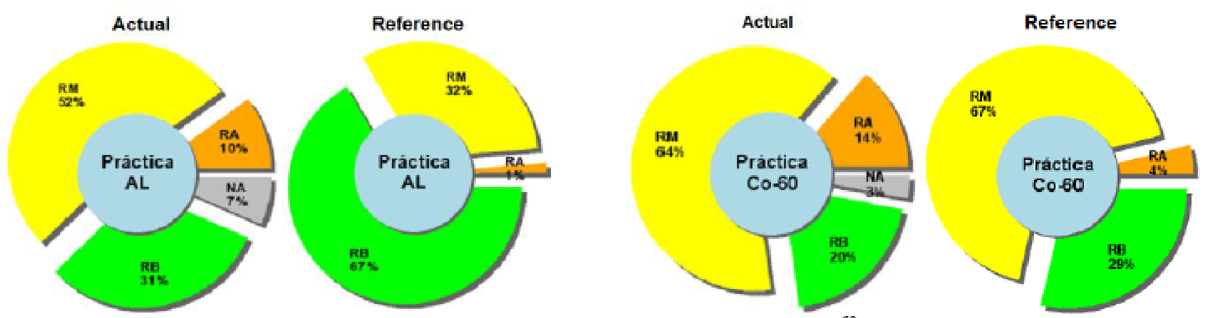
Radiation therapy, within medical applications, constitutes the highest risk practice due to the complexity of its operations and patient treatments. Taking into account, the historical analysis of the accidents that have occurred in this practice, where the occurrence of human errors and equipment failures is involved, it is necessary to carry out safety assessments to identify and prevent accidental exposures and provide staff with a tool and elements that allows them to manage risks. For this purpose, the effectiveness of the "risk matrix" method was known, which consists of a combined analysis of the frequency of occurrence of the accident-initiating event, the probability of human errors or failures of safety barriers and the severity of the consequences of the events, which allows defining criteria of acceptability based on risk.

This paper presents the Cuban experience in the application of this method in the safety assessment in radiotherapy services with linear accelerator (Linac) and teletherapy machine with cobalt-60 (^{60}Co). The work allowed identifying the most significant events that contribute to the risk, as well as the most appropriate action plan to reduce the probability of occurrence of events, strengthen security barriers and minimize the consequences. The risk of accidental sequences, applicable to the radiotherapy services studied, caused by possible human errors and equipment failures was estimated using the TOKSA computer tool, where it was found that none of the accidental sequences qualify as very high risk (RMA). In Linac machine high risk (RA) represents 10%, medium risk (RM) 52% and low risk (RB) shows 31%.



(a) Linac (b) ^{60}Co machine

Fig.1. Behaviour of risk analysis in radiotherapy: Actual and reference machine o services.

Keywords: Risk matrix, Linac, ^{60}Co teletherapy machine

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PS4 (T4.2-0763)

Dosimetry of occupationally exposed individuals managing patients on PET/MRI and PET/CT: a comparison study

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The advent of the PET/MRI technology has expanded the boundaries of investigation in nuclear medicine, supported by the high sensitivity of solid-state PET detectors [1]. Nonetheless, the coil positioning leads to an increased exposure period of the worker to the injected patient. This procedure does not occur on PET/CT and, therefore, exposure period reduced on such scanner. The aim of our study was to evaluate the dosimetry of two occupationally exposed individuals (OEI) working at the Center of Nuclear Medicine of Hospital das Clínicas of the University of Sao Paulo. We used thermoluminescent (TLD) dosimeters as monitors in pulse, thorax and crystalline on both PET/MRI and PET/CT during five months of clinical and research routine. We also monitored the time for positioning/removing the patient on both scanners.

For this study, OEI1 performed 76 PET/MRI studies and 102 PET/CT studies while OEI2 performed 26 and 56 PET/MRI and PET/CT studies, respectively. The mean±standard deviation is shown in Table 1. Despite the mean values of exposure on PET/CT were slightly higher than PET/MRI, it was not statistically relevant in all instances of organ evaluation (thorax, pulse or crystalline). The high values of standard deviation found was due to the relevant variation of doses along the months of evaluation (ex: doses on the pulse of OEI1 for PET/CT ranged from background(ground-scale) until 0.33 mSv). The average time of both workers for positioning the patient was 2.56±1.60 minutes and 9.55±9.98 minutes for PET/CT and PET/MRI, respectively. The slightly higher values of dose on PET/CT might be explained by the higher number of examinations. Our study encourages future investigations on the nursing staff, which is a critical population that might be exposed to ionizing radiation, mainly on dynamic studies, due to the synchronized injection must be with the protocol starting.

Table 1. Mean and average of the working months of each OEI for PET/MRI and PET/CT.

	OEI1			OEI2		
	PET/MRI	**PET/CT**	**p-value**	**PET/MRI**	**PET/CT**	**p-value**
Thorax	0.13±0.10	0.15±0.13	0.58	0.05±0.00	0.21±0.24	0.08
Pulse	0.10±0.05	0.31±0.33	0.19	0.08±0.03	0.25±0.27	0.18
Crystalline	0.09±0.05	0.23±0.23	0.36	0.09±0.06	0.26±0.30	0.36

Legend: OEI-Operational exposal individual, data are shown in the table as mean±standard deviation

Keywords: *PET/MR, PET/CT, dosimetry*

ACKNOWLEDGMENTS

The authors thank the biomedical staff for the collaboration to be monitored in our study.

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PS4 (T4.2-0766)

Development of application to visualize the spread of scattered radiation in radiography using Augmented Reality

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Purpose: In radiation medical care at a medical institution, many medical workers such as nurses in a ward are involved in radiography in a hospital room. Many medical professionals are busy and have few opportunities to get education on radiation protection. An understanding of radiation protection is necessary for medical staff to carry out their work with confidence against radiation. Understanding the behavior of scattered radiation is important for reducing exposure, but imaging the behavior of scattered radiation is difficult because the scattered radiation is invisible. Therefore, we developed an application that uses augmented reality (AR) to visualize the spread of scattered radiation during patient room radiography. Method: Using the Particle Heavy Ion Transport code System (PHITS), one of the Monte Carlo codes, we simulated the behavior of scattered radiation during patient room radiography using a portable X-ray system. Reproduced the patient with a voxel phantom that mimics the human body. The calculation results were converted to polygon data using ParaView, a three-dimensional visualization software, for each arbitrary relative dose divided by the ambient dose equivalent divided by the incident surface dose. Based on the polygon data, an application for tablet devices was created using Unity, a game development engine. AR content was created with ARkit. By tapping anywhere on the tablet camera, the behavior of scattered radiation spreads in three dimensions. Result: Using AR, the behavior of scattered radiation could be observed while the operator moved from an arbitrary angle. Conclusion: This application is useful for those who have no knowledge of radiation protection to intuitively understand appropriate protection methods by distance and shielding at the time of shooting in the room.

***Keywords**: Monte-Carlo simulation, Augmented Reality, Radiation protection education*

PS4 (T4.2-0768)

Average Annual Effective Doses of Personnel of the Republic of Belarus

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In Belarus, the Unified State System for Monitoring and Accounting for Individual Doses of Citizens is functioning. The collection of data on individual doses of personnel is carried out annually in the form of statistical reports that are received by the State Dosimetry Register. According to the State Dosimetry Register, an analysis is made of the number of institutions using sources of ionizing radiation, the number and average annual effective doses of personnel in dynamics over the period 2000 – 2017.
The analysis showed that the number of institutions using sources of ionizing radiation over the period from 2000 to 2017 increased significantly, almost 1.5 times (from 661 to 942), and mainly due to medical institutions (80%), which increased 1.7 times.
The number of personnel for the period from 2000 to 2017 increased 1.5 time. The total number of personnel of organizations that submitted reports for 2017 to the State Dosimetry Register amounted to 10,464 people, of which 72% are personnel of medical facilities whose radiation doses were obtained according to individual dosimetry data control.
The increase in the number of personnel in general and medical institutions, in particular, is explained by an increase in the number of medical diagnostic devices, as well as with the development of new methods of radiation diagnostics and in this connection an increase in the number of medical procedures associated with the use of ionizing radiation sources. The number of personnel in medical institutions increased 1.6 times, and the number of personnel in industrial, educational and other institutions increased only 1.4 times.
The average value of annual effective doses of external exposure to personnel of institutions of the Republic of Belarus using sources of ionizing radiation for the period from 2000 (3.22 mSv) to 2017 decreased by 3.2 times. The average annual effective doses of external training for medical staff (2.42 mSv) decreased 2.2 times over 18 years. At the same time, we note that since 2000 to 2008 the average effective dose to personnel decreased by 4.2 times, but since 2009 by 2017 increased by 2 times. The average annual effective dose of external exposure to personnel of industrial, educational and other institutions (4.02 mSv) decreased by 4.3 times over 18 years. Moreover, since 2000 to 2008 the average effective dose to personnel decreased by 3.2 times, and since 2009 to 2017 decreased by 1.5 times.
Thus, the radiation doses to the personnel of all institutions have significantly decreased for 18 years, despite the fact that since 2008 a slight increase has been revealed. The decrease can be explained by the improvement of methodological support and control in the field of radiation safety.
The observed “fracture” in 2008 - 2009 due to the use of more sensitive individual dosimeters. Since 2013 the doses for the personnel of both industries practically coincide within the limits of the average error.
While the number of personnel in medical facilities is on average almost 2.5 times higher than the personnel of industrial and other institutions, average doses are 1.6 times lower. The average annual individual effective dose of anthropogenic exposure to personnel in 2017 was 1.02 mSv. The radiation doses to personnel for 18 years are quite low and average 1.45 mSv per year, which is significantly lower than the dose limit for personnel. Personnel of industrial, educational and other institutions make the main contribution to the dose of anthropogenic exposure to personnel.

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Poster Session T4

PS4 (T4.2-0835)

Radiation exposure of staff in brachytherapy of eye tumors

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Brachytherapy of malignant eye tumors is performed using half-shell applicators (plaques) made of silver sewn onto the eyeball. The beta radiation from Ru-106 / Rh-106 (activity ~25 MBq, $E_{\beta,max}$ = 3.5 MeV) embedded in the silver plaque (Fig. 1) is used for therapy and is on the inside only 0.1 mm Ag shielded to the eye. In outward direction 0.7 mm Ag absorbing 95% of the electrons.
The aim of this work is getting an estimation of the occupational dose and the hand part body dose of the staff handling directly the applicators or which comes close to the patients. The exposed staff can be divided into two groups: 1. Persons who handle the plaques in order to apply them to the eye (operating room staff, cleaning and disinfection staff), 2. Physicians and caregivers are nursing the patients during the few days of plaque application. (essentially a few short contacts a day to apply eye drops or to cleanse the eye).
The first group is at risk of being exposed to beta radiation (range in air several meters).
These persons are monitored with standard personal dosimeters (i.e. dosimeter films). In addition, it happens to the surgeons that in individual cases the plaque does not have to be handled with tweezers but directly from the back with the fingers. Direct dosimetry is difficult here. However, dose estimation via local dose measurements and via dose calculations, however, result in very small values under the existing handling conditions.
If the plaque is attached to the eye, the eye ball absorbs the electrons; the generation of Bremsstrahlung in the low Z tissue of the vitreous body is small. So, for the second group with only a small number of patient contacts per year dose values of significantly less than 1 mSv/a result. The results are in agreement with the outcome of the work of Busoni [1].

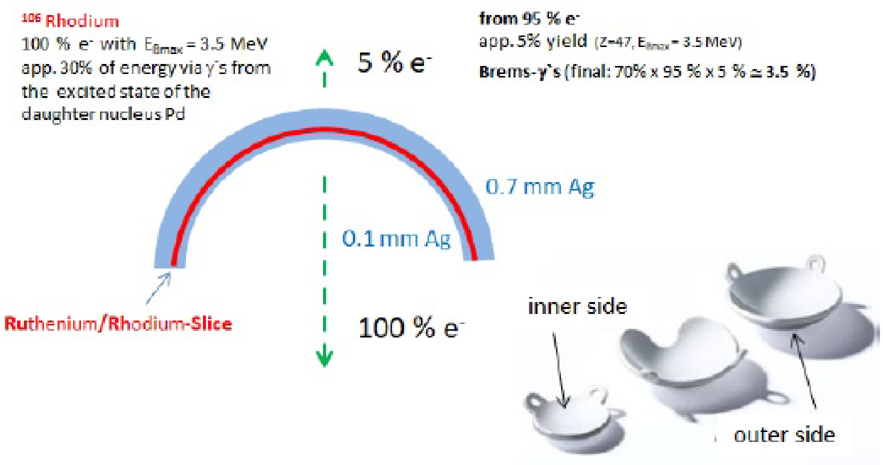

Fig. 1. Sectional drawing of an applicator; at right: 3 applicator designs in 3D

Keywords: Eye Tumor, Brachytherapy, Occupational Dose

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PS4 (T4.2-0857)

Percutaneous structural cardiology: are anaesthesiologists properly protected from ionizing radiation?

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Fluoroscopy guided interventions offer the possibility of performing minimally invasive surgery with less complications and faster patient recovery than open surgery. Percutaneous interventions sometimes need the use of high patient doses and it is necessary to protect both, patients and professionals from radiation risks. Anaesthesiologists, in most cases are supposed to stand at a safe distance from the patient, and this distance is far enough to keep their occupational doses close to background values. If anaesthesiologists have to stay close to the patient, the occupational doses might be higher. This can be the case of the transcatheter aortic valve implantation (TAVI) and other percutaneous cardiological structural procedures, usually performed in old patients with other risks, needing a more intensive patient care by anaesthesiologists. This work aims to investigate the radiation dose received by anaesthesiologists during this kind of procedures to improve their radiological protection.

Methods: Occupational radiation doses were measured prospectively during procedures using i3 electronic dosimeters (Raysafe, Sweeden) designed to measure Hp(10). Electronic occupational dosimeters were worn by professionals over the protection apron at chest level in addition to passive dosimeters worn under the apron. An electronic dosimeter was located at the C-arm, about 75 cm from the C-arm isocentre as a reference to measure the level of scatter radiation. Procedures were performed in one interventional laboratory in a university hospital equipped with a C-arm model Allura Clarity (Philips Health Care, Best, The Netherlands).

Results: Occupational doses have been measured in a sample of 49 procedures and main results are presented in table 1. The average dose per procedure received over the protection apron during TAVIs by the anaesthesiologist was 130 µSv, with a maximum value of 690 µSv. With these dose levels, an anaesthesiologist could participate in around 150 procedures before reach the regulatory annual dose limit for the lens of the eye in Europe (20 mSv).

Table 1: Occupational doses measured in µSv

	TAVI					Other structural procedures				
	N	Max.	Mean	Std. dev	P75	N	Max.	Mean	Std. dev	P75
C-arm	30	2530	586	435	745	15	3290	593	856	639
Anaesthesiologist	33	684	130	165	231	16	690	123	182	197

Conclusions: In those clinical procedures where patients need close anaesthesiology care, the anaesthesiologists might receive relevant radiation doses increasing their risk for cataracts. The proper use of occupational dosimetry will help to identify these situations. The use of protection glasses and a mobile shielding barrier is recommended to reduce radiation exposure to acceptable levels.

Keywords: *interventional cardiology, anaesthesiology, radiation protection.*

ACKNOWLEDGMENTS: This work has been partially funded by the Spanish Ministry of Economy and Competitiveness and European Development Fund (ERDF) under the project MEDICI number PI16/01413.

PS4 (T4.2-0873)

Analysis of scattering rays and shielding efficiency through lead shielding for 0.511 MeV gamma rays

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Radiation causes radiation hazards in the human body. World Health Organization classifies radiation as a carcinogen. When using radiation, maximum profit should be obtained from radiation, as per the as low as reasonably achievable principle, and thorough safety management is required to prevent damage due to exposure to radiation. In Korea, a case of radiation necrosis occurred in 2014. In this study, the scattering rays and shielding efficiency according to lead shielding were classified into epidermis and dermis for 0.511 MeV used in nuclear medicine. The ICRU provides various phantoms for radiation measurement. In this study, experiments were conducted using the slab phantom that represents calibration and the dose of human trunk. It is recommended to additionally insert a skin thickness of 2 mm when an ICRU slab phantom is used as the human trunk. In this study, the 2.062 mm thickness of the thorax skin was used as the insert, as recommended by ICRP 89, instead of 2mm skin thickness.

Experimental results showed that the shielding rate of 0.25 mmPb was 180% in the epidermis and 96% in the dermis. Shielding at 0.5mmPb showed shielding rate of 158% in the epidermis and 82% in the dermis.

As a result of measuring the absorbed dose by subdividing the thickness of the dermis into 0.5 mm intervals, when the shielding was carried out at 0.25 mmPb, the dose appeared to be about 120% at 0.5 mm of the dermis surface, and the dose was decreased at the subsequent depth. Shielding at 0.5 mmPb, the dose appeared to be about 101% at the surface 0.5 mm, and the dose was measured to decrease at the subsequent depth. As a result of experiments on the elimination rate of the scattering rays using the air layer between the lead shield and the phantom. It was determined that the scattering rays reaching the dermis with a thickness of 2 cm or more of the air layer were eliminated when the 0.25 mm lead shield was used. For the epidermis, an air layer of 4 cm or more was needed. When 0.5 mmPb shields were used, scattering rays reaching the dermis with a thickness of 1 cm or more in the air layer were eliminated, and an air layer of 3 cm or more was required for the epidermis.

High-energy photons mostly pass through without interacting with the human body, and Compton scattering is a major interaction. The scattered electrons improve the ratio of photons absorbed in the body, and an element with a higher atomic number, such as lead, results in a higher electron generation efficiency. Thus, the energy spectrum of gamma rays after lead shielding is changed by the braking radiation generated by the electrons. However, the changed spectrum (scattering rays) is not measured when the radiation dose is evaluated. The effects of scattering rays caused by wearing a lead apron cannot be ignored in the nuclear medicine department, which deals with many types of radiation isotopes.

This result suggests that when lead aprons are actually used, the scattering rays would be sufficiently eliminated due to the spaces generated by the clothes and air, Therefore, the scattering rays generated from lead will not reach the human body. The ICRU defines the epidermis (0.07), in which the radiation-induced damage of the skin occurs, as the dose equivalent. If the radiation dose of the dermis is considered in addition, it will be helpful for the evaluation of the prognosis for radiation hazard of the skin.

Keywords: PET, Scattering rays, Shielding, Lead

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PS4 (T4.2-0889)

Radiation Protection Education using Virtual Reality by Visualization of Scatter Distribution in Radiological Examination

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Purpose: When working on radiology and patient assistance in medical facility, radiation workers need to understand how to properly protect scattered radiation. In this study, we examined a visualization method to make it easy to understand the spread of scattered radiation in radiography room CT room and angiography room, and proposed its application to radiation protection education.
Methods: We constructed the X-ray radiography room, X-ray CT room, and angiography room using Particle Heavy Ion Transport code System (PHITS), and simulated the scattered radiation distribution when the patient was irradiated with X-rays. The three-dimensional distribution of each moment was continuously displayed to create a four-dimensional distribution. Using the created data, we conducted radiation protection education including exercises to make the students confirm the scatter distribution from any direction. The effectiveness of the scattered radiation visualization data was evaluated by a questionnaire.
Results: The position of assistance for standing chest radiograph was less scattered radiation at the side and below the patient. As a result of the questionnaire, this education has confirmed the effect of attracting attention about radiation protection. The visualization allowed students to understand the behavior of radiation and the source of scattered radiation.
Conclusions: Visualization of three and four dimensional scattered radiation distribution in the radiological examination room can intuitively enhance the understanding of the invisible radiation spread and appropriate aids.

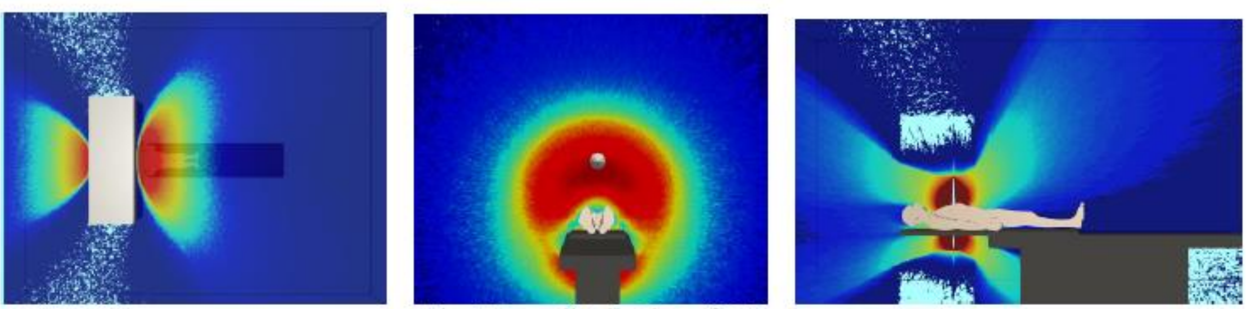

Fig.1 Dose distribution of CT room

Fig. 2 Dose distribution of angiography room

Key words : education and training; virtual reality; occupational exposure; monte calro simulation; four dimensional scatter dose distribution

PS4 (T4.2-0947)

Neutron measurements in the maze and outside of the operating room for different types of linear accelerators

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Radiation therapy linear accelerators (LINAC) working at energies higher than energy threshold for (γ, n) nuclear reaction of some materials irradiated by the photon beam, produce measurable number of neutrons. These LINACs have the capacity to produce photoneutrons in the target, flattening filters and collimating devices if operated at energies above 10 MeV. The neutron component in treatment rooms where photon energies larger as 10 MV are produced is significant. To protect staff at radiotherapy departments, bunkers with maze are designed having thick primary and secondary shielding.

In recent years, Hungary has received a large number of new accelerators most of which were placed in already existing bunkers. The aim of our work was to measure the neutron dose for the different types of LINACs in Hungary. The study was performed with 14 linear accelerators of 11 radiotherapy centers for 15 MV and 18 MV photon energies; the accelerators are: Varian (TrueBeam, TrueBeam HD and Novalis), Siemens (Primus and Artiste) and Elekta (Synergy and Versa). Measurements of neutron fields were conducted at the beginning part (A) and at the end of the maze (B) with Berthold LB 6411 neutron dose rate meters (Fig.1.). The results show that the neutron field depends on the construction of the LINAC head and the photon beam energy. Further examinations are necessary to study the effect of the shape of the maze on the neutron field.

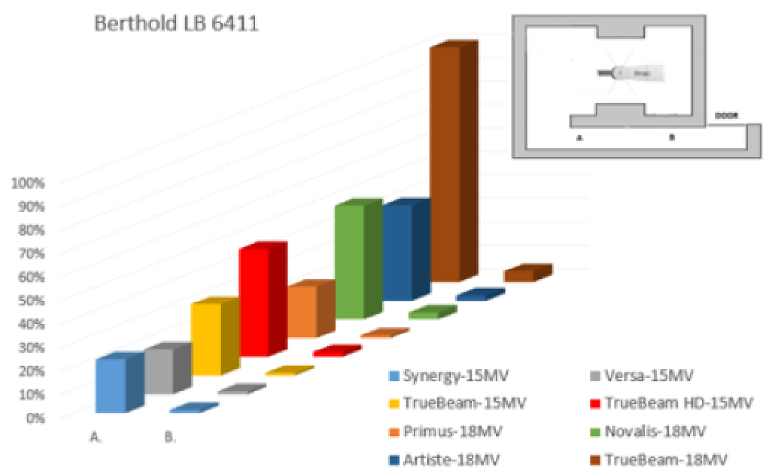

Fig. 1. The comparison of neutron doses for different LINACs (15 and 18MV photon beam energies) expressed in percentages.

Keywords: *radiotherapy, neutron, linear accelerator*

ACKNOWLEDGMENTS

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PS4 (T4.2-P0973)

Brain CancEr risk in joint cOhort of MEdical workers exposed to Ionizing radiation in France and Korea (BECOME)

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Context
Cancer is one of the main radiation-induced diseases that can occur from exposure to low doses of ionizing radiation (IR), as shown in many epidemiological studies. Among medical professionals, increases in cancer risk have been observed for protracted exposures to IR. Some publications have reported an excess of brain tumor in interventional cardiologists. However, the link with the profession has never been formally proven, partly because of poor reconstruction of the doses received, often based on the career history of the practitioners.

Objectives
The objective of the **BECOME** (Brain CancEr risk in joint cOhort of MEdical workers exposed to Ionizing radiation in France and Korea) project is to conduct an international nested case-control study to quantify the radiation-induced risk of death from central nervous system (CNS) cancer, in function of the IR occupational doses received in three cohorts of medical professionals exposed to IR in France, USA and Korea.

Methods
The study is based on three national cohorts aiming at analyzing the risk of radiation-induced diseases in medical professionals: the O'RICAMs (Occupational Radiation-Induced Cancer in Medical Staff) cohort, set up in 2011 by the French Institute for Radiological Protection and Nuclear Safety (IRSN), follows 227,000 individuals since 2002, the USRT cohort, set up in 1982 by the National Cancer Institute, includes 110,000 radiotechnologists followed since 1926, and the Korean cohort, set up in 2017 by the Korea University and the Korea Center for Disease Control and Prevention (KCDC), includes 94,000 individuals followed since 1996. In 2014 and 2015 respectively, 40, 193 and 16 deaths from CNS tumors were respectively reported in the French, US and Korean cohorts. Five controls by case matched on year of birth, gender, and country will be recruited. For France and Korea, a complementary collection of information (including occupation, specialty, positions, alcohol/smoking status, and other co-morbidities) from occupational health records will be performed for the cases and the controls. The dosimetric reconstitution will be based on the prospective dosimetric monitoring of professionals provided by the SISERI (Ionizing Radiation Exposure Monitoring Information System) database for the French cohort and by the centralized national dose registry maintained by the KCDC for the Korean cohort. Organ doses have already been calculated for the US cohort. Conditional logistic regression methods will be used to estimate the relationship between CNS tumor risk and the occupational cumulative dose of IR, taking into account potential confounding factors.

Expected results
The inclusion of 249 cases and 1245 matched controls will allow detecting at least an odds ratio of 1.7 for a statistical power of 87% and an alpha significance level of 5%. The study will quantify the relationship between occupational exposure to low doses of IR and CNS tumor mortality risk in medical workers Our international joint approach could be extended to assess the dose-response for other type of cancers and non-cancer outcomes

Conclusion
These results will increase the knowledge on health effects of protracted low-dose exposures to IR and will help to improve radiation protection standards for professionals, but also for the general population exposed to low doses of IR.

PS4 (T4.2-1054)

Training radiation exposed workers in Dutch hospitals: A complete toolkit

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The 2013/59/Euratom guideline states that all individuals involved in the practical execution of medical radiological procedures must have received adequate training and education for the purpose of medical radiological practices, as well as relevant competence in radiation protection. However, it is not stated in which manner and how often these trainings should occur. Across Dutch hospitals, these training programs and their occurrence therefore differ vastly.

Since both the number and types of radiological procedures are rapidly increasing, there is a need to provide each radiation-exposed worker with an efficient training (and retraining) program. Due to diversities in tasks, the background and learning preferences of these workers, it remains challenging to provide a fitting training programme for everyone.

We have developed a complete toolkit for the continuous training of radiation-exposed workers in Dutch hospitals. For the development of this toolkit, we evaluated the current training curricula in six Dutch hospitals. Additionally, we observed radiological procedures and conducted structural interviews among different stakeholders such as radiologists, radiation protection officers and technologists. We also tested the added value of digital solutions such as e-learnings and more innovative solutions such as interactive augmented reality apps as supportive learning tools, taking into account the individual needs and learning objectives for different groups of health care professionals.

Classical sessions meet the need of the workers to obtain an overview of all the material, while being able to interact with the radiation expert to clarify difficulties. On-the-job training provides workers with a chance to implement the learning goals into daily practice and becoming more consciously aware during radiological procedures. In addition, the utilization of digital solutions like an e-learning allows workers to view the material in their own time and pace. We found that a combination of these different teaching styles results in the most effective training to the exposed workers.

Fig. 1. An overview of the training toolkit on the learning methods for exposed hospital employees.

Keywords: Training, Staff, Radiation

PS4 (T4.2-1176)

Radiation dose distribution of the surgeon and medical staff on orthopedic Balloon Kyphoplasty in Japan

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Balloon Kyphoplasty (BKP), a new treatment for vertebral compression fractures, requires relatively short surgery time, but is performed under continuous fluoroscopy. Radiation exposure of Orthopedic surgeon, chiefly, fingers that cannot be avoided them in fluoroscopy are expected to be at high doses. However, few reports are available on evaluation of BKP radiation exposure in Japan. To investigate the radiation dose distribution of BKP, we conducted a descriptive research among orthopedic surgeon and medical staff. Table 1 shows the annual dose estimates of BKP surgery by orthopedist.

Table 1. Based on the results of this survey, estimation of cumulative radiation dose from orthopedic surgeon on BKP operation in FY2018

	Left ring finger (mSv)	Lens on the left side (mSv)	Neck on the left side (mSv)	Chest on the left side (mSv)
	Hp(7)	Hp(3)	Hp(10)	Hp(10)
23 cases[a]	50.37	8.27	3.91	1.15
28 vertebral bodies[b]	40.88	6.72	3.08	0.84

a. Estimated based on the number of cases
b. Estimated based on the number of vertebral bodies.

Keywords: *dosimeter for eye lens, dosimeter for fingers, Balloon Kyphoplasty*

ACKNOWLEDGMENTS
We thank Dr. Matsuzaki H, Mr. Kagayama S and Mr. Yoshida H in National Hospital Organization Disaster Medical Center for their coordination and supervision, data collection. We also appreciate Ms. Ichika and medical staff in National Hospital Organization Disaster Medical Center for study support.

PS4 (T4.3-0013)

Assessment of uncertainties in the positioning of patient for IMRT treatment using portal device

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The patient set up at the linear accelerator for intensity modulated radiation therapy at "*Clínica de radioterapia la Asunción*" is monitored using portal device. The systematic error was quantified during the patients positioning in the linear accelerator Varian 6EX for IMRT treatment for three months. The average values were determined in the positioning of patients for: a) prostate, b) head and c) breast.

Previously, the intrinsic error associated with the accelerator was determined, it was at a maximum value of 1.6 ± 0.7 millimeters for the longitudinal displacement. The maximum error associated with the patient set up for the breast was 4.8 millimeters; which is considered to be associated with the patient's breathing. The lowest displacement was obtained in the skull, with a value of 3.7 millimeters.

Table 1. Set up error in breast patient

Error (mm)	Lateral	Longitudinal	Vertical
Average systemic error	0.3	-2.6	-3.0
Systematic group error	4.8	3.6	4.6
Group random error	0.47	0.36	0.45

Keywords: Radiotherapy, Set up patient, IMRT

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PS4 (T4.3-0134)

POTENTIAL FOR CT DOSE REDUCTION BASED ON QA PHANTOM AND HUMAN CADAVER IMAGES

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Purpose: CT centres having higher radiation doses when compared to many CT centres locally and internationally have been identified through a national dose survey carried in Nigeria for CT examinations of the brain, chest and abdomen. This provides a basis for dose reduction methods to be explored. This study proposes a methodology using a quality assurance (QA) phantom and human cadaver images.

Materials and methods: The study proposes a methodology consisting of three phases. Phase I: Manipulation of scan parameters to monitor their effect on radiation dose indices and psycho-physical parameters using a GE QA phantom, leading to determination of the optimal parameter settings. Phase II: Application of the identified optimal QA phantom protocols on the human cadaver as a starting point for further optimisation, which was followed with analysis of results based on VGA, VGC and VGR from evaluation of cadaveric images. Phase III: Clinical implementation of the finally optimised protocols and further image quality evaluation based on VGA, VGC and VGR.

Results: Fifteen adult QA optimised protocols were established in five CT centres. Of the 15 QA phantom protocols, nine protocols from three centres were tested on cadaver as two of the centres did not give permission for cadaver scanning. Of the nine protocols, six were further optimised as three protocols could not be further optimised due to severe loss of image quality. Furthermore, two of the six cadaver optimised protocols, were not implemented due to the radiologists preference of image quality, rather the established QA optimised protocols were retained which still achieved dose reduction when compared to standard protocols. The implemented optimised protocols in two of the three CT centre showed evidence of dose reductions based on the new CTDIvol DRLs as follows: 36% for brain; 18% for chest and 21% for abdomen, whilst the DLPs were: 44% for brain, 54% for chest and 30% for abdomen with acceptable image quality based on the VGA, VGC and VGR.

Conclusion: The study established optimised protocols in selected CT centres providing acceptable image quality based on the VGA, VGC and VGR using a methodology with a lot of advantages such as low cost resources (QA phantom and human cadaver) which is ideal for countries lacking expensive equipment (Catphan and anthropometric phantoms).

Key words: computed tomography; cadaver, QA phantom; image quality.

PS4 (T4.3-0143)

Optimization of patient dose with a dose registration system

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In the European Basic Safety Standards (2013/59/Euratom) it is stated that equipment used for interventional radiology and computed tomography, installed after 6 February 2018, must have the possibility to transfer all relevant parameters necessary for assessing patient dose to the patient file. Under the Dutch legislation all CT's and modalities for interventional radiology have to have this possibility. Also all other devices emitting ionizing radiation that are purchased after 6 February 2018 have to have this possibility. To comply with this recent legislation Noordwest Ziekenhuisgroep has decided to use a so called dose registration system (DRS).

All modalities that are used in Noordwest send their relevant parameters to our DRS. The DRS calculates the effective dose and sends out a dose report to be used in the patient file. The information is automatically placed in the report that is written after the examination. This report is to be read by the referring practitioner. In this way the requirements in Dutch legislation are fulfilled and the referring practitioner gets more information and knowledge about the effective dose and risk of examinations.

The DRS is also used in Noordwest Ziekenhuisgroep to optimize the patient dose for different procedures for different modalities (e.g. CT, interventional cardiology, interventional radiology). In this way the exposure of the workers is also optimized.

In the presentation we pay attention to the implementation of the DRS and the first results of patient dose optimization are given.

Keywords: *Optimization, Patient dose, Dose registration system*

ACKNOWLEDGMENTS
We acknowledge all the colleagues at Noordwest, who helped with the implementation of the DRS and / or helped with the optimization of the patient dose for different modalities.

PS4 (T4.3-0472)

An alternative material for the patient radiation safety: Potassium Iodide and Hydroxyapatite

Andrea Vargas-Castillo, Angel M. Ardila

Lead is a material that has been used for many purposes for thousands of years. Due to its properties as density, ductility, stability and because it is relatively easy to extract and recycle, it is the most commonly used element as an effective shield against electromagnetic radiation (X and gamma rays) in radiological protection. Despite its excellents properties for the construction of radiation shielding, by producing a great absorption of incident radiation, its easy adaptation to the shielding and its very economical cost, it has been associated to effects harmful to health. Therefore, the main of this research is to propose a lead-free and non-toxic material that allows reduce the dose received by patients due to dispersion. For this purpose, the characterization of sintered hydroxyapatite (HAp) and potassium iodide (KI) was carried out, using different techniques such as X-ray Diffraction (XRD), X-ray Dispersive Energy (EDX), Fourier-transform infrared spectroscopy (FTIR) and Raman spectroscopy, to obtain information on the physical, chemical and structural properties of materials. On the other hand, the attenuation capacity of the X radiation was determined with an X-ray equipment (30-60 kV) and an RTMS (Real-Time Multiple Strip) detector, taking into account the intensity of the radiation as a function of the thickness and density of the sample compared to lead. In conclusion, it was identified that the non-metallic material that best attenuates the X-rays in this study for the possible application in radiological protection is potassium iodide.

PS4 (T4.3-0482)

Radiation risks assessment of medical exposure in Russia: current status of the problem and ways to solution

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The IAEA's Basic Safety Standards of 2014 require registrants and licensees to ensure that the patient has been informed, as appropriate, of the expected diagnostic or therapeutic benefits of the radiological procedure as well as the radiation risks. The same requirement is present in the Russian Basic Sanitary Radiation Protection Standards of 2009 [1] and in the Basic Sanitary Rules for Radiation Safety of 2010 [3]. It is the first time the requirements of estimating risk of late stochastic effects associated with medical radiologic exposure at the planning stage have been included in international and national regulatory documents. ICRP Publication 103, recommends evaluating radiation risks associated with diagnostic imaging using doses to the individual tissues at risk. Effective dose can be used for comparing doses from similar diagnostic technologies and procedures in different medical clinics and countries, as well as for comparing different technologies for the same examination if reference patient groups are similar by age and sex [3]. The impetus for the requirements was the increasing use of radiologic examinations, especially computed tomography (CT) and Positron Emission Tomography/Computed Tomography (PET/CT). Thus, the estimating of radiation risks and benefits as well as the Risk Communication in the cases of medical exposure are necessary. The "global" goal of the scientific research recent started in A. Tsyb MRRC is to develop a methodology for assessing the possible radiation health effects to patients when undergoing medical diagnostic procedures using radiation technology taking into account the ICRP mathematical risk models (Fig 1). The developing method for determination of radiation health effects when single and multiply pass CT examination based on estimates of lifetime attributable cancer risks is highly topical and may find practical application in the current system of radiological protection.

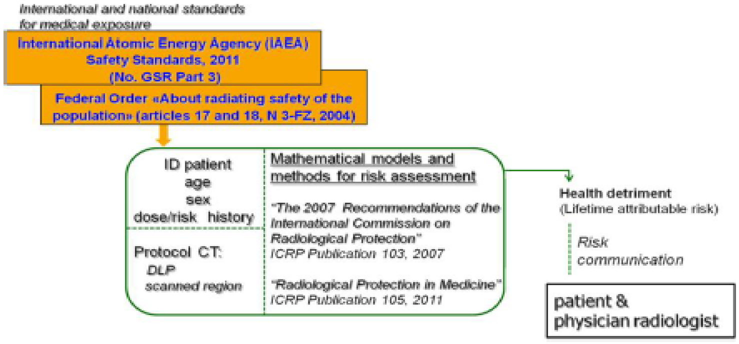

Fig. 1. Methodology of cancer risk estimates following diagnostic CT radiation exposures

Keywords: Medical exposure, Radiation risk, Radiological protection

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Poster Session T4

PS4 (T4.3-0610)

Proposed Diagnostic Reference Levels for Paediatric Plain Chest Radiography in Northeast Nigeria: A Multi-Centre Study

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Background: Paediatric patients constitute 10% of annual radiography examinations performed worldwide. Studies have shown that radiation doses received by paediatric patients can vary considerably from one hospital to another, even for the same type of examination indicating a need for optimization of doses consistent with the clinical objective.

Objective: To propose a Diagnostic Reference Level for paediatric plain chest radiography through measured entrance skin doses of radiation and correlating same with anthropometric parameters such as weight, Body Mass Index and anteroposterior chest thickness and technical parameters such as x-ray tube potentials (kVp) and tube load (mAs) in selected tertiary hospitals in Northeast Nigeria based on the current European Guidelines on Diagnostic Reference Levels for paediatric imaging.

Materials and Method: This was a cross-sectional study conducted in five (5) Federal Tertiary Hospitals in Northeast Nigeria. A total of 231 paediatric patients were recruited for this study. Patient demographic data such as age and gender; anthropometric data such as weight, height, Body Mass Index, anteroposterior chest thickness and technical exposure parameters such as tube potential (kVp), and tube load (mAs) were obtained. Calibrated thermoluminescent dosimeter chips were placed at the centring point of the patient's chest to measure entrance skin dose generated by the tube kVp and mAs. Median and standard deviation were used to summarise the data while percentiles were used to establish Diagnostic Reference Levels. Pearson's correlation was used to determine the relationship between doses received with anthropometric and technical parameters. Diagnostic Reference Level was set at the 75th percentile of the median patient dose distribution for the various age groups in all the centres studied. Statistical significance was set at $p < 0.05$.

Results: The results of the established Diagnostic Reference Levels for the various age groups in the five centres studied were; Centre A: 1Month - <4Years, 3.31mGy; 4Years - <10Years: 3.77 mGy; 10Years - <14Years: 4.07 mGy; 14Year - <18Year, 4.75 mGy; Centre B: 1Month - <4Year, 3.03 mGy; 4Year - <10Years, 1.48 mGy; 10Years - <14Years, 4.07 mGy; Centre C: 1 Month - <4Years, 1.02 mGy; 4Years - <10Year, 1.82 mGy; 10Years - < 14Year, 1.43; 14Years - <18Years, 1.89 mGy; Centre D: < 1Month: 1.94 mGy; 1Month- < 4Years, 2.91 mGy; 10Years - < 14Years, 2.35 mGy; Centre E: 4Years - <10Year, 1.56 mGy; 10Years - <14Years, 1.23 mGy; 14Years - <18Years, 2.12 mGy. Pearson correlation shows statistically significant ($p = 0.005$) strong positive relationship ($r = 0.940$) between kVp and entrance skin dose in Centre A.

Conclusion: Proposed Diagnostic Reference Level for paediatric plain chest radiography marked by variations in measured entrance skin doses was established for specific age groups of paediatric patients in this study. A statistically significant strong positive relationship between kVp and entrance skin dose was observed in centre A thus, indicating a need for optimization.

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Keywords; entrance skin dose, diagnostic reference levels, paediatric, chest radiography

PS4 (T4.3-0651)

Overview of Published Data on Diagnostic Reference Levels in South Africa

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INTRODUCTION

The objective of Diagnostic Reference Levels (DRLs) is to help avoid radiation dose to the patient that does not contribute to the clinical purpose of a medical imaging task. A DRL is usually set in a suitable reference group of patients, based on either weight or height. The aim of DRLs is to improve a regional, national or local distribution observed for a general medical imaging task, by identifying and reducing the number of unjustified high or low values in the distribution. Additionally, DRLs promote good practice for a more specific imaging task and also promote an optimum range of values for a specified medical imaging protocol.

South African legislation states that reference dose levels should be introduced for applications in diagnostic x-ray examinations.

METHOD

A literature search was performed to find DRL data specifically for South Africa.

RESULTS

A total of 12 publications were found, some in the form of published congress abstracts. These include three publications on Barium meals and/or Barium swallows and/ or Barium meals and dose reductions over time for these procedures. There were five publications on DRLs for selected fluoroscopically guided diagnostic and/or interventional procedures, as well as three publications on computed tomography DRLs and one for general X-ray procedures. The largest study included over 40.000 procedures.

CONCLUSION

Some published DRL data exists in South Africa, but it is rather patchy and uncoordinated. Certain hospital groups in South Africa have started implementing dose optimization programmes and awareness is increasing, especially with the formation of the South African chapter of AFROSAFE. A national DRL project was proposed to the IAEA and will start in 2020.

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PS4 (T4.3-0705)

Usefulness of ^{18}F-FDG PET/CT imaging for inflammatory lesion evaluation on aorta among the IgG4-RD patients

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Immunoglobulin G4-related disease (IgG4-RD) has an immune-mediated condition. The diagnose requires integration of features, such as fibroinflammatory changes, infiltration by IgG4+positive plasma cells into target organs and elevation of IgG4 serum level. Also, IgG4-RD can affect cardio-vascular system, and cause changes in large/medium-sized vessels. ^{18}F-FDG PET/CT is an important cancer management imaging tool, both for diagnosis and staging, treatment response and showing metabolic and functional condition related to disease. Recently, it has been found that FDG PET/CT be also useful for inflammatory lesion evaluation. The purpose of work is to evaluate the usefulness of PET/CT images using ^{18}F-FDG tracer for detecting inflammatory changes on aorta wall among histologically/clinically proved of IgG4-RD patients. We conducted a retrospective study of 13 elder patients (8 men and 5 women) with histologically/clinically proven IgG4-RD diagnose evaluated by PET-CT with ^{18}F-FDG tracer. Baseline data included in our study are listed below on Table1. FDG-PET maximum standardized uptake (SUVmax) and target to background ratio (TBR) were calculated. SUVmax is conducted by placing an individual region of interest (ROI) around the lesion on ascending aorta. Background was defined as the SUVmean of Left atrium (LA) and Spleen, and ROI were placed over them. Results showed that IgG4 serum level with SUVmax and TBR (reference: spleen) tend to show correlation which is very close to significant level (p=0.0528, p=0,0514). On the other hand, correlation between IgG4 serum level and TBR(LA) did not reach statistical significance (p=0.1349). The total number of cases was relatively small and abdominal aorta uptake of ^{18}F-FDG is considering to be analyzed in future. However, 18F-FDG is found to be useful for evaluation of IgG4-related disorder.

Table 1. Baseline Features of the 13 patients with histologically/clinically proven IgG4-RD

Variable	Mean	Std Dev	N
Age(y)	64.15	13.26	13
Weight(kg)	61.92	16.65	13
Height(m)	163.42	6.18	13
BMI (kg/m^2)	23.29	6.39	13
HR (Heart Rate)	71.69	8.82	13
Systolic BP (mmHg)	119.15	12.21	13
Diastolic BP (mmHg)	71.46	11.20	13
White blood cell (10^9/L)	6.18	1.62	13
Red blood cell (10^{12}/L)	4.20	0.68	13
Hemoglobin (g/dL)	13.09	2.08	13
Mean Corpuscular volume(fL)	93.97	4.65	13
Mean Corpuscular hemoglobin concentration (g/dL)	33.20	0.70	13
Glucose (mg/dL)	106.45	45.65	11
HbA1C(%)	15.36	28.14	8
LDL-Cholesterol(mg/dL)	118.25	32.23	4
HDL-Cholesterol(mg/dL)	46.33	9.22	6
Triglyceride(mg/dL)	118.17	66.74	6
Creatinin(mg/dL)	0.89	0.47	13
Blood urean nitrogen(mg/dL)	15.85	6.62	13
C- reactive protein(mg/dL)	0.53	0.82	13
Glomerular filtration rate	69.80	23.74	13
Immunoglobulin G4(mg/dL)	1000.23	962.18	13
Immunoglobulin G(mg/dL)	2330.50	717.09	6
C3 levels(mg/dL)	97.93	37.30	11
C4 levels(mg/dL)	20.20	11.54	11

Keywords: IgG4-related disease, 18F-FDG PET-CT, Inflammation

PS4 (T4.3-0719)

Optimization of therapeutic effects and patient comfort in thyroid cancer therapy using iodine-131

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Differentiated thyroid carcinomas (DTC) are the most frequent malignancies of the endocrine system in 2015 in Japan. The management of patients with DTC involves radioactive iodine-131 (RAI) therapy to complement the surgical resection of the thyroid gland or to destroy metastasis. Precautions are necessary to restrict the radiation exposure to public, to the patient's family and to the staff treating these patients. Number of guidelines and regulations, mostly based on administered doses, have been established when patients shall be discharged from hospital. The purpose of this study is to contribute for optimization thyroid cancer treatment by iodine-131 based on individual patient characteristics. Materials and methods: We conducted retrospective study including 87 cases with DTC who had received RAI from June 2016 till July 2018. Patients who had complete information on primary tumor size, serum TSH, T3, T4, Tg (one month before, on the day of the treatment, and one month after receiving RAI), as well as renal function and exposure rate were included in our study. Dose rate was measured (immediately, 12h, 24h, 36h, 48h, 60h) at 1 meter distance from anterior mid trunk using NaI scintillation survey meter. Using these dose rates, clearance rate of I-131 was calculated with simple model assuming I-131 clearance as mono-exponential using Microsoft Excel. Univariate and multivariate correlation analyses were used to compare factors which made influence on clearance of Iodine-131 using dedicated statistical software (JMP Pro14). P-values less than 0.05 were considered significant. Results: Univariate analysis showed significant associations of effective half-life with several factors (gender, body height, body weight, eGFR, T3 one month after receiving RAI). However, it did not show significant associations when those factors are analyzed at multivariate analysis. Only eGFR and serum T3 one month after receiving RAI tend to show significant associations. Conclusions: We consider that gender, body height, body weight, eGFR and T3 one month after receiving RAI are significant factors for clearance rate of radioactive iodine from the human body in DTC patients. Using these factors, doctors can predict clearance of I-131 and plan the date to release patients from restricted area. Effective management of these factors could help the doctors to optimize the occupancy of isolated rooms with higher output, but also reduce the operational cost of the hospital and improving the comfort to patients.

Keywords: thyroid cancer, Iodine-131, thyroid cancer therapy.

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PS4 (T4.3-0723)

Strengthening Knowledge and Skills in Radiotherapy Quality and Safety in Latvia

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Training on radiation protection and safety in Latvia is organized according to Cabinet Regulation No. 752 of 22 December 2015, which states the requirements of training in radiation safety issues within the scope of a course programme developed by an educational establishment not less than once every five years for employees working with ionizing radiation, both in medical and non-medical areas. This training is coordinated by the Radiation Safety Centre and the professional association of the relevant sector.
Currently, training on radiation safety issues in the medical area is provided by three higher education institutions in Latvia, but the training is aimed for medical staff like radiologists, medical physicists (MPs) and radiographers in diagnostic imaging, as it covers radiation safety issues at low radiation doses only. The Faculty of Medicine at the University of Latvia is the exclusive provider of the radiation protection and safety course modules in radiation therapy with mostly theoretical education within undergraduate studies for therapeutic radiographers or radiation therapists (RTTs). There is no specific training on radiation safety issues for practicing RTTs in Latvia: every five years, they have to undergo training on radiation safety issues for radiographers working with low radiation doses only. Thus, RTTs should also be trained on radiation safety issues at high radiation doses and dose rates.
The aim of the IAEA TC project LAT0003 "Strengthening Knowledge and Skills in Radiotherapy Quality and Safety in Latvia" was to train the trainers, upgrade existing equipment and to introduce new visual aids for training in radiation safety, thus improving knowledge and practical skills for RTTs. During the two and half years project there were involved 6 external experts, including 1 radiation oncologist, 2 medical physicists and 3 RTTs. Only 2 RTTs and 13 trainees from other areas were able to participate in the train-the-trainers course, however the national training course had a great interest from practicing RTTs. During the project two RTTs had the observership at radiotherapy center in the Netherlands and one trainer – 2 weeks observership in UK. The equipment for treatment planning, patient immobilization and treatment verification has been procured and is planned for implementation into training of practicing RTTs, undergraduate and postgraduate students in radiotherapy. The UL and Riga Technical University have agreed on the use of radiotherapy training equipment in the process of medical physicists training.

Keywords: *radiotherapy, therapeutic radiographers, radiation safety*

PS4 (T4.3-0756)

Personalized restrictions to the patient after a ^{177}Lu DOTATATE treatment

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Introduction: ^{177}Lu-DOTATATE radionuclide therapy is a source of public external exposure. The purpose of this study was to retrospectively evaluate the impact of different social contact times assumptions on the period of restrictions.

Methods: 23 patients with a total of 59 treatment cycles have been included in this retrospective study. Dose rates had been measured at 1 m from the patient's anterior mid trunk, at 1, 4, 8, 24, 48 and 96 h after treatment administration. Effective half-life ($Teff_{1/2}$) was evaluated with the last 3 measurements. Patient release at 24 h after treatment was assumed. Restriction period were assessed using as input data the effective half-life and dose rate at 24 h specific for each patient. Different social scenarios were considered, with contact times from elsewhere [1]. Two conditions during the restriction period were compared: No contact (as in [1]) and short contact. Dose constrain per cycle and normal contact after the period of restrictions are summarized in table 1 for the different scenarios.

Results: Mean activity administered (± standard deviation (sd)) per cycle was 6946 ± 598 MBq. Dose rate after treatment administration and at 1, 4, 8, 24, 48 and 96 h were 33±5, 24±4, 17±5, 13±5, 9±4, 7±4, 5±3 µSv/h, respectively. Mean half-life was 3.05 ± 1.02 days. Table 1 shows the periods with restrictions considering no or minimal contact after patient release at 24 after the treatment. The increment of the period with restrictions is around 2 days, when minimal contact is allowed.

Table 1. Period with restrictions (days), with minimal contact and no contact, after ^{177}Lu-DOTA treatment

Scenario		At home		Child 0-2 y		Child 2-5 y		Child 5-11 y		General worker
Dose constrain/cycle		0.75 mSv		0.25 mSv		0.25 mSv		0.25 mSv		0.3mSv
Normal contact after the period of restrictions [1]		6 h @ 1 m 8 h @ 0.1 m		15 x 35 min @ 0.1 m		8 h @ 1 m 4 h @ 0.1 m		4 h @ 1 m 2 h @ 0.1 m		8 h @ 1 m
Contact during restrictions		6 h @ 1 m	No contact	3 h @ 1.5 m	No contact	8 h @ 1.5 m	No contact	4 h @ 1.5 m	No contact	No contact
Days with restrictions	Mean±sd min-max	7±6 0-29	6±4 0-19	12±7 2-38	10.±6 2-28	10±6 1-31	8±5 1-24	5±4 0-18	7±7 0-29	2±3 0-11
Difference	Mean±sd min-max	2±3 0-16		2±2 0-10		3±3 0-15		2±4 0-24		-- --

Conclusions: After ^{177}Lu-DOTATATE therapy, the increment of the period with restrictions is minimal when short contact is allowed with the family members after patient release, in comparison when no-contact is imposed. This minimal contact would be better accepted by the patient.

***Keywords**: Lu-177, Radiation dose, Restrictions*

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PS4 (T4.3-0762)

Learning from ionising radiation dose errors, adverse events and near misses in UK clinical imaging departments

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Since 2008, there has been a successful standardised method for classifying and reporting radiotherapy errors (RTE) and near misses within the United Kingdom (UK). All UK RT departments support the voluntary collection of anonymised RTE data to Public Health England (PHE).

A similar national standard reporting framework has been developed for UK Clinical Imaging departments for staff to use to code and report errors, adverse events and near misses. Under the auspices of the UK Clinical Imaging Board (CIB), a joint professional body working group was established to develop a framework – something that had never been done before!

The now published Coding Taxonomy details relevant categories, coded to fit both the UK Ionising Radiation (Medical Exposure) Regulations (IR(ME)R) duty holder roles and the relevant steps of the patient pathway and is to be used with the standard Reporting Template. A guidance document and User Guide are no at https://www.rcr.ac.uk/sites/default/files/cib-learning-from-adverse-events-user-guidance.pdf This includes a classification and pathway coding system intended to enable organisations to code, analyse and learn from errors. The CIB report provided a number of recommendations including the establishment of a multidisciplinary steering group to develop this work to a national level. Several UK imaging departments are already using this system. It is envisaged that the Regulator for IR(ME)R in England will also use this as part of their notification policy.

Over 29 million diagnostic, interventional and nuclear medicine examinations were delivered by, or for, the NHS in England in 2016. In the delivery of these large numbers of medical exposures, inevitably things can and do go wrong no matter how dedicated and professional the staff. Reviewing errors and near misses must be done locally to ensure learning takes place. It is also important that national reporting with anonymised data being shared is published for learning purposes across the UK.

The poster will update delegates on this CIB partnership work and the national implementation phase with support from the multidisciplinary steering group.

PS4 (T4.3-0804)

Analysis of Usage of Computed Tomography in Korea based on UNSCEAR Classification

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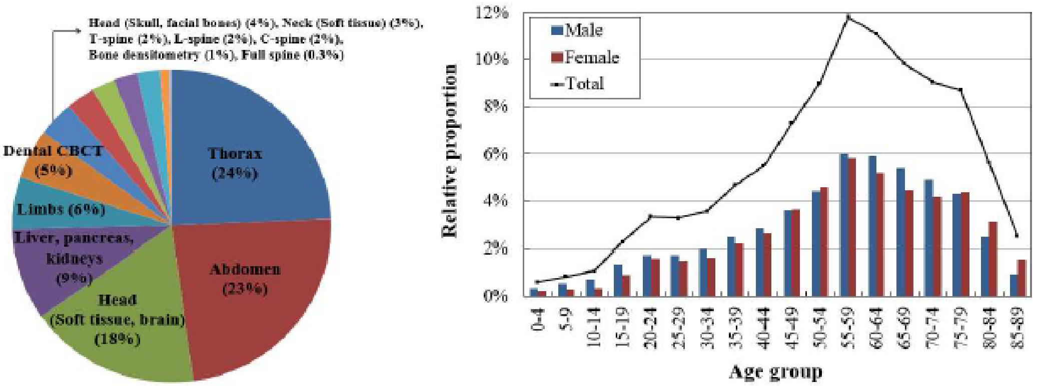
Computed tomography (CT) is used as an important tool in modern medicine, and its usage is rapidly increasing due to of its various application [1, 2]. In addition, the patient dose by CT is higher than other x-ray examinations. Therefore, the patient dose management by CT is important. UNSCEAR is an international organization for the analysis of the effects and risks of radiation on human body. The UNSCEAR collect information about medical radiation by each countries, and periodically publish reports based on data collected. Since Korea is a member of the United Nations, it is necessary to submit information to UNSCEAR. The objective of this study is analysis of usage of CT in Korea based on UNSCEAR classification to establish basic data for patient dose management and to provide information standardized to UNSCEAR. We analyzed classification system for medical radiation of UNSCEAR. UNSCEAR classify medical radiation into diagnostic and interventional radiology, nuclear medicine, and radiotherapy. CT is included in diagnostic and interventional radiology modality in the classification, and divided into 18 scan regions such as thorax and abdomen. We collected raw data for usage of CT in 2017 from Korea Health Insurance Review & Assessment Service. The data were analyzed by scan regions, gender, and age using SAS program, a commercial statistical analysis program. As a result, total usage for CT in Korea was about 9.4 million in 2017. Analysis by scan region showed Thorax (24%), Abdomen (23%), and Head (Soft tissue, brain) (18%) were high, and other were less than 10%. Analysis by age showed that the usage increased with age, reached the peak for 55-59 group, and then decreased. Analysis by gender showed that usage of male was higher at most age groups under 74 years, and usage of female were higher at 45-54 years and 75 years and older. The results of this study can be used as a representative data for usage of CT in Korea, and can be used as basic data for of medical radiation for international coordination.



(A) (B)

Fig. 1. (A) CT usage by scan region, (B) CT usage by age and gender

Keywords: Computed tomography, UNSCEAR, Medical radiation

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PS4 (T4.3-0824)

Estimation of the dose in the upper gastrointestinal series

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[Introduction]

The diagnostic reference level is used in the dose of radiodiagnosis. Although this supports X-rays CT, IVR, mammography, etc., sufficient information is not acquired in upper gastrointestinal series. In upper gastrointestinal series especially in a medical checkup, it is greatly influenced by a technologist's technology, a patient's bodily shape, etc. In this research, it aimed at estimating a dose for each radiography posture and conditions in the upper gastrointestinal series by a medical checkup.

[Methods]

In upper gastrointestinal series in a medical checkup, the exposure dose and the entrance surface dose were calculated using entrance surface dose calculation software (SDEC) from the voltage, current, and time for each radiography posture. Total fluoroscopy time was measured and the exposure dose and the entrance surface dose were calculated similarly.

[Results]

A calculation result is shown in Table.1. In supine position double imaging, the difference for each radiography posture became small. It is prone position full statue that the dose became the smallest, supine position 2nd in oblique position (the distributing method) and half standing supine position 2nd oblique position (Schatzki' position), the dose became high. As a result of total fluoroscopy time's having been about 60 seconds, and calculating fluoroscopy voltage by 70, 80, and 90 kV and calculating fluoroscopy current by 1.0 and 1.5 mA, the dose increased with the rise of voltage. The dose increased, so that current became high.

[Conclusion]

The exposure dose and entrance surface dose in upper gastrointestinal series have been estimated. The dose for each radiography posture needs to be recognized and irradiation conditions and exposure field size needed to be taken into consideration. It was confirmed to become dose reduction by shortening fluoroscopy time as much as possible.

Table 1. The exposure dose and entrance surface dose of each radiography posture

	Supine position double imaging	Prone position full statue	Distributing method	Schatzki' position
Exposure dose (C/kg)	5.50×10^{-6}	5.17×10^{-6}	7.28×10^{-6}	6.89×10^{-6}
Entrance surface dose (mGy)	0.406	0.381	0.540	0.511

Keywords: Exposure dose, Entrance surface dose, Upper gastrointestinal series, Entrance surface dose calculation software

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PS4 (T4.3-0832)

Performance evaluation of gamma-ray real-time imaging system with digital camera

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Extravasation occurs with a certain probability in nuclear medicine examination. By real-time imaging using a commercially available camera and scintillator, accumulation and contamination of RI can be easily confirmed, and the inspection can be quickly stopped when an abnormality occurs. The purpose of this study is the performance evaluation of the sensitivity and spatial resolution of a gamma-ray imaging device using digital cameras. The CsI scintillator was illuminated with X-rays, and the light emission was analyzed from live view images when using only PENTAX KP (RICOH, Tokyo), when two types of image intensifier, C9016 (HAMAMATSU, Shizuoka) and XX2050M (Nakanishi Image Lab. Lnc, Tokyo), were connected respectively, and when using a high-sensitivity CMOS camera, ORCA-spark C11440-36U (HAMAMATSU, Shizuoka). The relationship between the pixel value and the dose rate was evaluated by changing the dose rate of the X-ray irradiated to the scintillator. In the case of PENTAX KP alone, light emission could not be confirmed below 10 mSv / h, but light emission could be confirmed up to 0.38 mSv / h by using C9016. With ORCA-spark C11440-36U, light emission could be confirmed up to 0.06 mSv / h. The emission at the dose rate predicted by nuclear medicine inspection was confirmed by using a high-sensitivity digital CMOS camera.

Keywords: *Extravasation, real-time imaging, digital CMOS camera*

PS4 (T4.3-0903)

National diagnostic reference levels in interventional cardiology in Spain

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Aims: Following ICRP recommendations [1] and the European regulation [2], the national DOCCACI group (Spanish acronym for dosimetry and quality criteria in interventional cardiology) has proposed and updated national DRLs for coronary angiography (CA) and percutaneous coronary interventions (PCI). 14 hospitals collected patient dose indicators from 26 interventional rooms: kerma area product (KAP), kerma at patient entrance reference point (K_R), fluoroscopy time and number of radiographic (cine) images. Information about the dose rate at the phantom entrance in reference conditions (20 cm PMMA) for the X-ray units for the standard imaging modes was also provided.

Results: Table 1 shows the updated DRLs to be proposed to the national health authority compared with the DRLs proposed during the previous period 2009-2013:

Table 1. Updated Diagnostic Reference Levels (DRLs) proposed

Period	CA 2009-13	CA 2014-17	PCI 2009-13	PCI 2014-17
KAP (Gy·cm^2)	42	40	89	78
K_R (mGy)	-	532	-	1319
Fluoro time (min)	6,7	6,7	15	15
Images number	780	727	1300	1270

Important differences between centres were found. The median values of KAP in each centre ranged from 12 to 53 Gy·cm^2 for CA and from 28 to 134 Gy·cm^2 for PCI. Median values of fluoroscopy time for each centre, ranged from 3 to 8 minutes in CA and from 7 to 20 minutes in PCI. When the X-ray dose rate at the entrance of the phantom (20 cm PMMA) in standard conditions were compared, the values ranged from 3.0 to 15.4 mGy/min for fluoroscopy in low dose mode, from 7.6 to 29.1 mGy/min for fluoroscopy medium, from 10.1 to 50 mGy/min for fluoroscopy high dose mode and from 0.040 to 0.22 mGy/image, in cine.

Conclusions: New national DRLs have been proposed for interventional cardiology in Spain. Large differences between centres in median values of KAP and fluoroscopy time show that there is a wide margin for optimization. Dose rate measurement at the entrance of the phantom, in reference conditions, also shows important differences. The optimization process should continue.

Keywords: Interventional cardiology, patient doses, diagnostic reference levels, optimization.

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PS4 (T4.3-1043)

The secretome of irradiated non-tumorigenic mammary cells MCF-10A elicits DNA damage in MCF-7 and MDA-MB-231 breast cancer cells

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Radiotherapy is used as an adjuvant treatment post-surgery in breast cancer because some foci of cancer cells may linger at the remaining breast tissue or the mastectomy site. It was long accepted that the lethal effects of ionizing radiation occur as a result of damage to the DNA in irradiated (IR) cells. However, DNA damage response mechanisms are activated and may promote cell survival with efficient repair or with genomic alterations. Chromosomal aberrations are frequent in surviving cells and may enhance the chromosomal instability (CIN) which has been associated with increased risk of recurrence and metastasis. Besides, scientific evidence indicates that intercellular communication can affect the response in IR cells and even cause damage in non-irradiated (N-IR) cells. In this study, we evaluated the effect of the secretome of non-tumorigenic mammary epithelial cells (MCF-10A) on proliferation and biological responses related to DNA damage in breast cancer cells (MCF-7 and MDA-MB-231). Our findings showed that conditioned media from IR and N-IR MCF-10A cells produces cycles of DNA double-strand breaks in N-IR and IR tumor cells leaving them with residual damage. CIN markers (micronuclei, nucleoplasmic bridges and nuclear buds) were also increased in both IR and N-IR tumor cells, being the effect of conditioned media from IR MCF-10A greater in many cases. Besides, the clonogenic survival of tumor cells was differentially modulated by conditioned media from MCF-10A: it was decreased in MCF-7 and enhanced in MDA-MB-231 cells. These results evoke the relevance of tumor-host interaction in tumor behavior and the response to radiotherapy.

Keywords: *DNA damage, Secretome, Irradiation, breast cancer*

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PS4 (T4.3-1056)

Irradiation induces mesenchymal traits in PANC-1 and BxPC3 pancreatic adenocarcinoma that are hindered by the antihistamine ranitidine

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In a study performed previously in our laboratory it was demonstrated that a 2 Gy irradiation increased the growth of human pancreatic PANC-1 xenografts in nude mice, while slowered growth of BxPC3 grafts. A 2 Gy irradiation also increased the development of lung metastasis in both tumor types. On the other hand, the antihistamine ranitidine (R) hindered the growth of PANC-1 and BxPC3 tumors and the development of PANC-1 lung metastasis in nude mice.
This study aimed to evaluate the effect of irradiation and R on epithelial to mesenchymal transition (EMT), a process associated with invasion and metastasis, in pancreatic tumors.
Dedifferentiated PANC-1 and more differentiated BxPC3 tumors were irradiated (I) or not (C) with 2 Gy of gamma radiation, transplanted to non-irradiated mice, and treated with R 150 mg/kg.day, p.o. (I+R; C+R) or not (I; C). Immunohistochemistry was performed to evaluate the expression of EMT molecular markers (E-cadherin, vimentin, Slug) and TGF-β1 (a major promoter of EMT). In PANC-1 tumors the epithelial marker E-cadherin was not detected in any group, while transcription factor Slug nuclear expression (mesenchymal marker) was similar in all of them. In C-grafts we observed a big number of vimentin (mesenchymal marker) and TGF-β1 positive cells that was even bigger in I-tumors, but not in R and I+R. In BxPC3 only I-tumors did not show E-cadherin at cell membrane in the inner areas of slices. Very few cells expressed vimentin and TGF-β1 in C-tumors; this expression was enhanced in I-group but not changed in I+R or R. An increase in nuclear Slug and TGF-β1 was detected only in I-grafts. TGF-β1 correlated positively with vimentin in both tumor types. Nuclear Slug positively correlated with vimentin and TGF-β1 only in BxPC3.
In order to perform in vitro studies, PANC-1 and BxPC3 cell cultures were pretreated (R) or not (C) for 1 day with R 20 μM, then irradiated (I; I+R) or not (C, C+R) with 2 Gy of gamma radiation. R treatment continued for 5 days; then, indirect immunofluorescence, Western blot or migration experiments were carried out. Results showed an increase in vimentin expression, intracellular TGF-β1 and cell migration in both I cell lines that was blocked by R. The transcription factor Slug showed the same behavior as in vivo experiments in both cell lines. In BxPC3 cells, E-cadherin was highly expressed in C, reduced in I, and partially re-expressed in I+R. C-PANC-1 cells expressed very low levels of E-cadherin that disappeared in I cells.
In conclusion, our results showed that a dose of 2 Gy of gamma radiation induced a gain of mesenchymal features characteristic of EMT in both human pancreatic tumor cell lines which emphasize the complexity of tumor response to radiotherapy. Also, R treatment reduced this effect pointing out the relevance of research on drugs that control both growth and metastasis.

Keywords: *Pancreatic adenocarcinoma, epithelial to mesenchymal transition, irradiation.*

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PS4 (T4.3-1060)

Appropriate frequency of computed tomography scan for setup before intensity modulated radiation therapy

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Intensity modulated radiation therapy (IMRT) is highly accurate treatment modality which needs delicate radiation delivery. For the accuracy, physician usually use variable tools for fixing patients, positioning setup, and verification of real position. Pre-treatment computed tomography (CT) scan is one of most important verification tool for setup and many physicians performed CT scan before every treatment delivery. However, the effectiveness of daily CT scan has not truly proved yet, and excessive exposure of radiation also has not been justified. Thermoplastic mask was usually used to fix the position of treatment target in treatment of patients with head and neck cancer. Inter-treatment setup error was significantly decreased by the mask, so that the need of pretreatment CT scan for setup may be decreased. This study investigated the setup errors, weekly volume changes of target volumes and organs-at-risk, and resulting dosimetric changes IMRT for head and neck cancer patients.

We analyzed the pretreatment CT images of 28 patients with head and neck cancer who treated with volumetric IGRT at Seoul St. Mary's Hospital, Seoul, South Korea. Total number of CT scans were 90 weekly scans in 15 patients and 403 daily scans in 13 patients. Setup error was evaluated with auto-registration on adaptive targeting software, and Volumetric changes were measured in repeated CT scans every week. Dosimetric changes were also re-calculated in repeated CT scans. Every process was confirmed by 3 radiation oncologist.

There showed no statistical differences in positioning difference between weekly and daily CT scan (Table 1). Though significant volume changes of gross target volume and parotid gland were occured during treatement, which did not cause dosimetric changes to each target.

In conclusion, daily CT scan is not necessary for accurate treatment delivery. Weekly scan is enough for IMRT and can spare excessive radiation exposure to the patients.

Table 1. Positioning differences between weekly and daily CT scan

	Mean positioning error (mm, 95% CI)			
	Total (n=493)	Weekly CT(n=90)	Daily CT(n=403)	p value
Lat.-Med.	1.91 (1.77-2.05)	1.81 (1.55-2.08)	1.95 (1.79-2.11)	0.393
Ant.-Post.	1.64 (1.51-1.76)	1.72 (1.51-1.92)	1.61 (1.46-1.76)	0.393
Sup.-Inf.	1.45 (1.34-1.55)	1.62 (1.39-1.85)	1.38 (1.27-1.50)	0.443
3-Dimension	1.67 (1.60-1.74)	1.72 (1.58-1.85)	1.65 (1.56-1.73)	0.070

Keywords: *radiation therapy, intensity modulated radiation therapy, image guided radiation therapy*

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Poster Session T4

PS4 (T4.3-1085)

Comparison of Voluntary Incident Reporting and Learning System

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Purpose
Voluntary incident reporting and learning systems have been utilized to improve safety and quality for radiation therapy by reporting incidents and near misses and by sharing these experience within the community There are well-known voluntary incident reporting and learning systems, RO-ILS (Radiation Oncology Incident Learning System) from ASTRO (American Society for Radiation Oncology) with the support of the AAPM (American Association of Physicists in Medicine), ROSEIS (Radiation Oncology Safety Education Information System) from ESTRO (European Society for Radiotherapy and Oncology), SAFRON (Safety in Radiation Oncology) from IAEA (International Atomic Energy Agency). In this study, we compared these three systems to investigate the differences between these systems.

Materials and methods
RO-ILS, ROSEIS, and SAFRON systems were compared with information about the system, which was obtained by searching the homepage, other websites, and literature of RO-ILS, ROSEIS and, SAFRON to the extent that we could access. We also referred to the information that can be informed on the homepages of the ASTRO, ESTRO, and IAEA which manage the systems. It was focused on that background, operating structure, data security, feedback, the benefits of participating in each system, and these were compared to each other.

Result
RO-ILS, ROSEIS, and SAFRON are similar to systematic aspects. ESTRO and IAEA offer the system by their server. However, RO-ILS was externally made and managed by Clarity PSO (Patients Safety Organization), and ASTRO is served as a gateway. The reviews uploaded on the homepage that analyze the reported data are provided to the participated institution, but the format of feedback was different in each system. RO-ILS provides institution-specific summary reports to each institution and uploads aggregate data reports to the homepage, including summary report cards and case reviews. SAFRON provides a newsletter that includes interesting news about radiotherapy and SAFRON reports. ROSEIS provides the function that users can make the database, analyses and custom searches available at the site, and it also sends newsletters including spotlight cases to institutions. The main difference is whether benefits to participation exist. There are Medicare's Merit-Based Incentive Payment systems (MIPS) in the United States and participation in RO-ILS can meet the requirement of MIPS. Furthermore, participation in RO-ILS offers benefit to the maintenance of the board. However, there is no specific benefit except review report in ROSEIS and SAFRON. This difference of benefit may affect the amount of participation. While ROSEIS got 1074 reports and SAFRON got 1657 reports in its first 8 years, 7968 reports were submitted to RO-ILS in its first 4 years.

Conclusions
These systems are operated through voluntary reporting by participating institutions, and the reported data is shared confidentially with a community without punishment. Since RO-ILS is a national system that shares the same legislation and medical structures, it can provide benefits for participation, and these financial benefits may attract more institutions to participate. This active participation would eventually promote a safety culture.

Keywords: Voluntary reporting and learning system, Patient safety, Radiotherapy

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PS4 (T4.3-1130)

The MARRTA Project, Application of the Risk Matrix approach to Advanced Techniques in Radiotherapy: A multidisciplinary effort to raise awareness involving the patients to improve safety in Radiotherapy

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The MARRTA project (Risk Matrix for Advanced Techniques in Radiotherapy) presents the continuation of a successful approach (MARR project) to develop a National Program on risk analysis with the involvement of all parties, to raise awareness of risk and provide the patient with high safety and quality treatments in the radiotherapy processes involving advanced techniques (IMRT / IGRT / SBRT).

The risk matrix methodology, developed by the Ibero-American Forum of Radiological and Nuclear Regulatory Agencies [1-2], was used to build a theoretical risk model of the radiotherapy process, analyzing all potential human errors and equipment failures and their associated barriers. As part of the MARR project [3], the original model was adapted to the Spanish 3D conformal radiotherapy practice, a risk matrix application guide was written to facilitate the application of this risk analysis methodology in any Spanish radiotherapy service and several training courses were conducted by the MARR coordinating group.

The realization that advanced radiotherapy techniques (IMRT, IGRT, SBRT) are becoming the standard treatment of choice, lead the "Forum of Radiation Protection in healthcare settings" (CSN and Professional Societies) to start the MARRTA project, with the objectives of further extending the risk model to include these techniques, releasing a new software tool that offers enhanced usability and flexibility, and developing a public information campaign aimed at the patients, to provide them with recommendations and information, and to promote their active participation in the quality and safety of their treatments.

Keywords: *Risk Analysis, Radiotherapy, Risk Matrix*

ACKNOWLEDGMENTS

Partially funded by research grant by Fundación Mapfre (Ayudas Investigación Ignacio H. de Larramendi).

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PS4 (T4.3-1213)

Analysis of Interventional Procedure Usages in Korea

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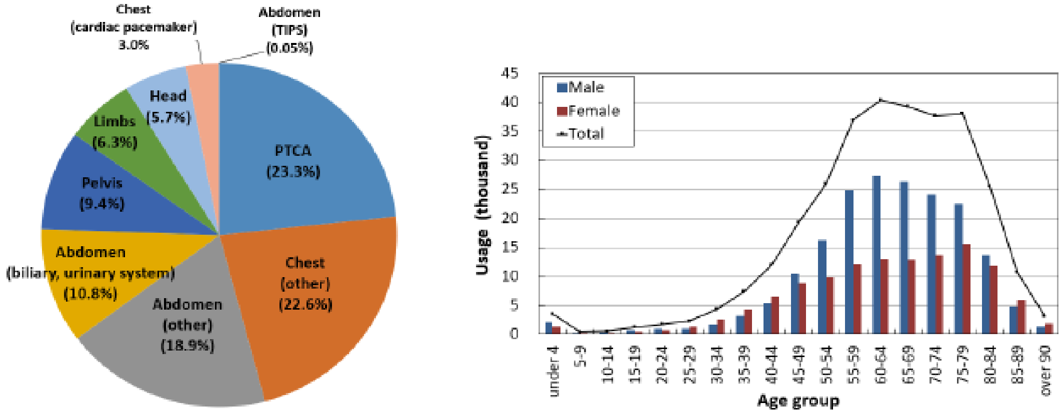
Interventional procedures are an important tool in modern medicine for treatment of vascular or non-vascular diseases. The usage of interventional procedures has been increasing over time due to the development of health industry and the public interest in health [1, 2]. Interventional procedures require long fluoroscopy time due to their complex and diverse procedure methods, resulting in higher patient doses compared to other diagnostic radiology. Therefore, radiation safety management by interventional procedures is required at national level. The objective of the present study was to analyze interventional procedure usages in Korea. We collected raw data of the number of interventional procedures performed in 2017 from Korea Health Insurance Review & Assessment Service (HIRA). The data were analyzed using statistical program SAS. Total 114 interventional procedures were classified into 9 groups by specific procedures and examination region on body such as PTCA (Percutaneous transluminal coronary angioplasty), TIPS (Transjugular intrahepatic portosystemic shunt), head, chest, abdomen, pelvis, and limbs. Based on the established classification, the usage data were further analyzed by patient gender and age. The patient ages are grouped by every 5 years. The total number of interventional procedures was 0.33 million in 2017. By procedure types, usage was generally high for PTCA (23.3%), chest (other) (22.6%), and abdomen (other) (18.9%). By gender, male (60%) took more interventional procedures than female (40%). By age, usage of interventional procedures was increased with age until 60-64 age group. After that group, the number of interventional procedures decreased with age. The result of this study can be used for patient dose management by interventional procedures and radiation safety of medical radiography.



(A) (B)

Fig. 1. (A) Interventional procedures usage by procedure type, (B) Interventional procedures usage by gender and age

Keywords: Interventional procedures, Usage, Medical radiation

ACKNOWLEDGMENTS

This work was supported through the KoFONS using the financial resource granted by NSSC. (No. 1803013)

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PS4 (T4.3-1218)

Dose reconstruction in dental cone-beam computed tomography

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There is an increasing call for radiation dose tracking from medical examinations and the patient-specific dose management has become a great concern [1]. It is highly desirable that the radiation dose and potential cancer risk associated with each computed tomography (CT) scan of each patient be reported and documented to manage the patient dose in CT examinations. However, the average dose measurement with a standard-size cylindrical phantom is the only method for the practical dose estimation in CT examinations. One of the other alternatives to estimate patient-specific dose distributions is the Monte Carlo (MC) method, but it requires extremely high computational cost, although they can provide accurate estimates [2]. The other methods for estimating patient-specific doses generally employ several approximations due to the complexity of photon transport in heterogeneous media.
The authors have developed a numerical approach to estimate the patient-specific dose distributions in the cone-beam CT. The developed algorithm is based on the ray-tracing techniques and requires the reconstructed voxel data in values of linear attenuation coefficients and the scanning protocol. The algorithm can be divided into two parts. First, the algorithm calculates the absorbed dose due to the primary photons by considering beam attenuation along the beam path between the source and each reconstructed voxel in conjunction with the solid angle subtended by given voxel. Then, from the pre-calculated primary dose value in a given voxel, the scatter dose values to all the other voxels are similarly calculated as the primary dose. The developed algorithm shows a good agreement with the MC simulation for the anthropomorphic phantom. The accuracy of the analytical method is investigated by comparing results with the MC estimates and the acceleration strategy using deep-learning techniques is also discussed.

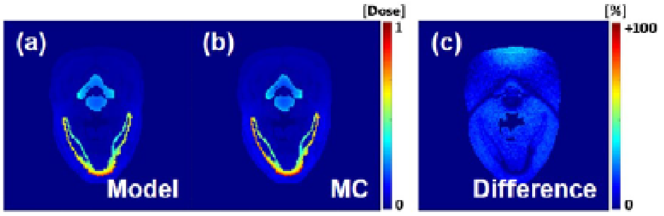

Fig. 1. The spatial distributions of the absorbed dose in the anthropomorphic phantom. (a), (b), and (c) represent the results from developed algorithm, MC simulation, and their relative difference, respectively.

Keywords: Cone-beam computed tomography, Patient dose, Dose distribution

ACKNOWLEDGMENTS
This work was supported by the National Research Foundation of Korea (NRF) through the Korean government (MSIP) under Grant 2017M2A2A6A01071267.

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Poster Session T4

PS4 (T4.3-1220)

Estimation of clearance rate of Iodine-131 from thyroid cancer patients based on individual patient's characteristic

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Differentiated thyroid carcinomas (DTC) are the most frequent malignancies of the endocrine system in 2015 in Japan. The management of patients with DTC involves radioactive iodine-131 (RAI) therapy to complement the surgical resection of the thyroid gland or to destroy metastasis. Precautions are necessary to restrict the radiation exposure to public, to the patient's family and to the staff treating these patients. Number of guidelines and regulations, mostly based on administered doses, have been established when patients shall be discharged from hospital. The purpose of this study is to contribute for optimization thyroid cancer treatment by iodine-131 based on individual patient characteristics. Materials and methods: We conducted retrospective study including 87 cases with DTC who had received RAI from June 2016 till July 2018. Patients who had complete information on primary tumor size, serum TSH, T3, T4, Tg (one month before, on the day of the treatment, and one month after receiving RAI), as well as renal function and exposure rate were included in our study. Dose rate was measured (immediately, 12h, 24h, 36h, 48h, 60h) at 1 meter distance from anterior mid trunk using NaI scintillation survey meter. Using these dose rates, clearance rate of I-131 was calculated with simple model assuming I-131 clearance as mono-exponential using Microsoft Excel. Univariate and multivariate correlation analyses were used to compare factors which made influence on clearance of Iodine-131 using dedicated statistical software (JMP Pro14). P-values less than 0.05 were considered significant. Results: Univariate analysis showed significant associations of effective half-life with several factors (gender, body height, body weight, eGFR, T3 one month after receiving RAI). However, it did not show significant associations when those factors are analyzed at multivariate analysis. Only eGFR and serum T3 one month after receiving RAI tend to show significant associations. Conclusions: We consider that gender, body height, body weight, eGFR and T3 one month after receiving RAI are significant factors for clearance rate of radioactive iodine from the human body in DTC patients. Using these factors, doctors can predict clearance of I-131 and plan the date to release patients from restricted area. Effective management of these factors could help the doctors to optimize the occupancy of isolated rooms with higher output, but also reduce the operational cost of the hospital and improving the comfort to patients.

Keywords: *thyroid cancer, Iodine-131, thyroid cancer therapy.*

ACKNOWLEDGMENTS
We gratefully thank to the staff of RI section of Nagasaki University Hospital.

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PS4 (T4.3-1241)

Assessment of Effective Dose, Renal Dose and Biological Risks during Abdominal CT Scans for Adults in a Moroccan Hospital

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Objective: This study aimed to estimate renal effective dose during abdominal, abdominopelvic, chest-abdominopelvic and urography CT scans in order to assess the renal risks of cancer and heredity per procedure in a Moroccan hospital.

Material and Methods: This study consisted of examining a total of 100 patients at the rate of 25 per localization. All CT examinations were performed with a GENERAL Electric 16 CT with Automatic Exposure Control System. The data that collected for each diagnostic exam chosen included scanner acquisition parameters, number of series, use of the contrast medium, and rotation time as well as slice thickness, the displayed CT dose index (CTDIvol) and the Dose Length Product (DLP). Renal dose, effective dose and biological risks were estimated using the International Commission on Radiological Protection (ICRP) conversion factor.

Results: The renal dose, the overall cancer risk R_{CR} and the risk of genetic effects R_{GE} in future generations per procedure were 16,96 mSv, 93 per 10^5 procedures and 34 per 10^6 procedures respectively at Abdomen CT. For abdomen-pelvis CT, there were 14,34 mSv, 79 per 10^5 procedures and 29 per 10^6 procedures respectively. Those at chest abdomen-pelvic CT were 15,26 mSv, 84 per 10^5 procedures and 31 per 10^6 procedures respectively. For urography CT, there were 17,07 mSv, 84 per 10^5 procedures and 34 per 10^6 procedures respectively.

Conclusion: Our values are relatively higher than those of published in some previous studies. Cancer risk and heredity estimation highlights the need to limit radiation dose. This first ever survey of CT practice in Hassan II Hospital of Agadir confirmed the need to improved training of health professionals involved in computed tomography on factors affecting image quality and dose and protocols optimization.

Keywords: *Abdominal CT scans, effective renal dose, cancer and heredity risks.*

PS4 (T4.4-0004)

Major Challenges to Achieving an Effective Assessment Process of Radiotherapy Services

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In accordance with the Law no. 111/1996, the National Commission for Nuclear Activities Control (CNCAN) is the national competent authority that exercises regulation, licensing and control attributions in the nuclear field, ensuring the peaceful use of nuclear energy and the protection of public and workers from the harmful effects of ionizing radiation.

Radiation therapy is a safe, effective and well-established way to treat cancer. It uses highly precise doses of radiation to kill cancer cells while avoiding irreparable damage to the surrounding healthy tissue. Advances in radiation therapy use the latest research in biology and physics and combine it with cutting-edge technology to deliver successful treatments.

The rapid growth in radiation treatment methodology is thus followed by an increasing need for appropriate training of radiation oncologists, medical physicists, radiologists and radiation technologists.

In Romania the number of radiation therapy services delivered by both public and private sector providers have increased considerably over the last 3 years.

The main challenge for CNCAN in both the review and assessment and authorization processes is the lack of professional staff, namely medical physicists and experts required for the practice of radiotherapy under the International Basic Safety Standards.

In the Norms of Radiological Safety on Radiotherapy Practices are mentioned requirements for the holder that shall ensure appropriate staffing levels to support the initiation of new services as well as the expansion or upgrade of existing services.

They should develop a plan to train the personnel and the training should be completed before installation of the equipment. According to the national standards the medical physicist specialized in radiotherapy physics is responsible for the following: dosimetry, radiation safety, treatment planning, quality control (QC), equipment selection.

Unfortunately, during the faculty the physicists don't have access to a postgraduate training in radiation oncology and clinical training in radiotherapy physics, therefore they need to work for two years in a radiotherapy centre in order to gain experience and become qualified medical physicist, and after five years of practical experience after qualifying in clinical radiotherapy physics, upon examination by the Ministry of Health and CNCAN according to the Norm of expert in medical physics they are recognized as medical physics experts in radiotherapy.

During the last year, some of the faculties of physics initiated training programs in radiotherapy physics undertaken in a hospital for the physicians under the supervision of an experienced or senior radiotherapy physicist.

Transition from simpler to advanced technologies does not necessarily result in a pro rata increase in staffing.

Use of the staffing algorithm requires a certain level of understanding of radiotherapy practice because of the amount of detail required in inputting relevant parameters. This is a highly dynamic environment, however, and therefore a lag in evidence is to be expected.

***Keywords**: radiotherapy, medical physicist*

PS4 (T4.4-0020)

Establishing Local Diagnostic Reference Levels (LDRL) for a Typical Fluoroscopic Examination in some Radiological Imaging Institutions in Ghana

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The primary aim of this research work was to propose the establishment of local diagnostic reference levels for typical fluoroscopic examination in some radiological imaging institutions in Ghana. Prior to the starting of this research a series of Quality Control Tests were performed using the Piranha kit to assess the whether the machine output is exactly what is expected. The research was done at two public hospitals (coded Facility A and B), located in the Greater Accra Region, between December 2017 and June 2018. A prospective quantitative research method was adopted to obtain frequency of fluoroscopic examinations in the Radiology Department [1]. The KAP (Kerma- Area Product) meter with all associated electronics was placed perpendicular to the central beam axis and in a position to completely intercept the entire area of the X-ray beam [2]. KAP and screening time values of the fluoroscopy procedure performed were obtained from the machine's console after each patient's examination. However, no additional adjustments or scan protocols were used for this research, to ensure the study reflected the actual normal practices in all the centres. The data obtained was statistically analysed using Microsoft Excel and results presented in descriptive statistics. The Diagnostic Reference Levels (DRLs) was estimated for each facility using the 75% percentile. A total of one hundred and thirty-six (136) patient dose data was collected for this study. DRL was established for the most frequently performed procedure which is hysterosalpingogram (HSG) examination and results compared with studies done elsewhere

Table 1. Comparison of Diagnostic reference levels (DRLs) of KAP values for HSG for this study with a study done in Kenya

Examination (HSG)	DRL values/ Screening Time
Facility A	6.0 $Gy.cm^2$/0.60 minutes
Facility B	4.1 $Gy.cm^2$/0.50 minutes
Wambani et al, 2014 Kenya Study	3.0 $Gy.cm^2$/ 2.1 minutes

Generally, the KAP values estimated as the DRL in this study were not deviating much from other studies. The Kenyatta National Hospital [3] had higher screening time value but lower KAP value than that of Facilities A and B by a factor of 3.5 and 4.2 respectively. Due to observed variations in KAP values, this work suggested standardization of protocols across facilities as a means to increase optimization of doses.

Keywords: Kerma- Area Product (KAP), Diagnostic Reference Levels (DRLs), Hysterosalpingogram (HSG)

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PS4 (T4.4-0322)

Reference based Simulation Study of Detector Comparison for BNCT-SPECT Imaging

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To investigate the optimal detector material for prompt gamma imaging during boron neutron capture therapy, in this study, we evaluated the characteristic regarding radiation reaction of available detector materials using a Monte Carlo simulation. Sixteen detector materials used for radiation detection were investigated to assess their advantages and drawbacks. The estimations used previous experimental data to build the simulation codes. The energy resolution and detection efficiency of each material was investigated, and prompt gamma images during BNCT simulation were acquired using only the detectors that showed good performance in our preliminary data. From the simulation, we could evaluate the majority of detector materials in BNCT and also could acquire a prompt gamma image using the six high ranked-detector materials and lutetium yttrium oxyorthosilicate. We provide a strategy to select an optimal detector material for the prompt gamma imaging during BNCT with three conclusions.

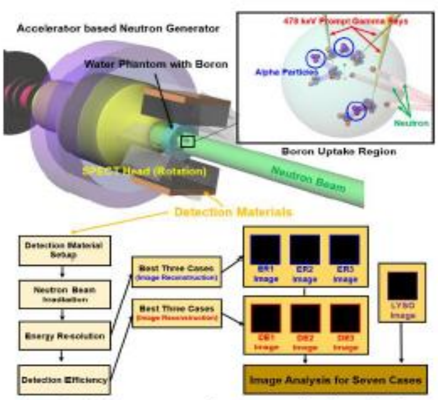

Fig. 1. Diagram summarizing the simulation procedure and simulation configuration. After changing the detection material, detection efficiency and energy resolution were determined for each detection material based on simulation results. The prompt gamma imaging was progressed using simulation information of only seven detection materials (three detection materials regarding each high performance of energy resolution and detection efficiency, LYSO).

Keywords: Boron neutron capture therapy (BNCT), Prompt gamma, Detector materials, Monte Carlo simulation.

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PS4 (T4.4-0549)

Radiomics and Radiation Dose Optimization

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Radiomics is a new advanced imaging processing approach to quantify medical images for new image biomarker discovery and further clinical decision-making. As a multi-step process, radiomics involves image acquisition, image processing, image segmentation, feature extraction/selection and data modelling. Studies have indicated that image acquisition is the first and main step for radiomics studies and challenges in this issue has a great impact on the radiomics feature values. To have more precise and accurate radiomics models, the high quality medical images are required which needs to change imaging protocols. In this light, many radiation dose may be delivered to the patients. To resolve this problem, new dose optimization process is suggested. In the present study, we aim to suggest a versatile radiation protection for radiomics data modelling. The main suggested issues are 1) to use low dose protocol, 2) image post processing, and 3) to use high robust features.

Keywords: *Radiomics, Radiation dose, Optimization, Robustness*

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PS4 (T4.4-0585)

Radiation protection study in Intraoperative Radiotherapy: calculation of peripheral dose around a *Mobetron* using Monte Carlo simulations

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Intraoperative radiotherapy (IORT) consists of external irradiation during a surgical intervention and is currently one of the most used treatments against cancer. Despite the shielding structures that protect the beam, there is a dispersed dose component that escapes from the accelerator head. The purpose of this work is to study the scattered radiation that is produced by the intraoperative linear accelerator Mobetron, which was installed in the hospital San Jaime in Torrevieja (Alicante).

One of the main problems is the scarcity of radiological protection studies useful in the phases of legalization of equipment. Therefore, this work aims to offer a useful tool for future users that allows the estimation of the dispersed dose in a simple way, as well as evaluation of the facilities and the necessary shielding to maintain the safety of professionals and hospital public.

To carry out this task, an approximate model of the linear accelerator head was designed, and after meshing the model, detailed simulations were carried out using the Monte Carlo code MCNP version 6. This study demonstrates the possibility of transferring the use of the Monte Carlo method to radiation protection studies in the field of radiotherapy. After calculating the disperse dose, the results were compared with experimental data available in the bibliography.

Theoretical results agree with experimental values, which shows that this method is a useful way to study the dispersed dose in the vicinity of a linear accelerator.

Keywords: *Dispersed dose, intraoperative linear accelerator, Mobetron, intraoperative radiotherapy, IORT, MCNP6*

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PS4 (T4.4-0740)

3DCRT Dose shaping for gynecological cancers with inguinal lymphatic chain nodes involvement: Technical aspects

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Accurate delivery of high radiation dose to a defined target volume with curative intent while minimizing exposure to surrounding healthy tissues in three dimensional conformal radiotherapy (3DCRT) is largely dependent on the advantages of computer based planning, treatment setup verifications and treatment record systems. These advantages are the reasoning for developing technical capacities.

In the study, technical guidelines for dose shaping in a 3D conformal forward planning for gynecological cancers with inguinal lymphatic chain nodes involvement is described, and validated with the classical three field (3-FLDs) and conformal (4-FLDs) planning techniques.

The 'Dose shaping technique' (DST) potentially minimizes bladder, bowel bag and rectal toxicities and allows for dose escalation in boost treatments or brachytherapy. This technique is feasible, especially with the use of beam-shaping static multileaf collimator (MLCs) segments which also serve in shielding out organs at risk (OARS).

Eleven cervical and ten prostate cancer cases with inguinal lymphatic nodes invasion previously planned and treated with either 3-FLDs or 4-FLDs were re-planned with the 'DST' to investigate superior conformity as well as optimum OAR sparing. Analysis of generated Dose Volume Histograms (DVH) based on a 46Gy per 23 fractions for both 3-FLDs and 4-FLDs showed dose coverage around 92.21% and 94.24% respectively, compared with 95.88% for DST. From DVH data, OAR sparing in terms of maximum/minimum doses was superior with the DST. An average dose homogeneity index of 1.08 was observed with the DST comparable to 1.23 and 1.31 for 4-FLD and 3-FLD respectively.

Keywords: inguinal, dose-shaping, toxicities

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The author expresses sincere gratitude to the oncology team and the research department of the Sweden Ghana Medical Centre for the diverse efforts towards this study.

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PS4 (T4.4-0741)

Minimizing cardiac dose in 3-Dimensional Conformal radiotherapy for left-sided breast cancer

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Dose optimization in left-breast cancer radiotherapy is mostly confronted with potential cardiac morbidity and mortality owing to anatomical position and motion. While the breath-hold and gating techniques are the gold standard in most places for minimizing cardiac dose, and improving coverage to the internal mammary nodes with 85% to 95% isodose, it is usually challenged by patient's discomfort and treatment verification during treatment. In a free breathing 3-D conformal radiotherapy forward treatment planning, advances are made to reduce cardiac doses by full optimization of available resources.

In this study, cardiac doses are compared using the multiple segmented field-in-field (FiF) technique and the traditional two tangential wedged-field (TwF) in a 3DCRT forward planning workspace using 6MV beams in a supine breast setup. It is anticipated that a formidable alternative cardiac dose reduction strategy shall be established, while exploiting the optimum potential of FiF for an improved dose homogeneity.

10 left-sided breast cancer cases previously planned and treated with the two TwF were re-planned using FiF. Cardiac dose reduction was achieved in the TwF by moving the med-tangential jaw of each field to cover the heart volume to less than 2cm. In the FiF however, MLCs in the first opposing subfields were drawn to shield out both the heart and lung volume. These subfields were weighted appropriately to exploit the full effect of these shields. With a 50Gy/25Fx prescription, the plans were normalized to the isocentre in the FiF, while in the TwF, this was a point within the PTV volume which generated adequate dose coverage. Analysis of average dose ($D_{average}$) and Maximum dose (D_{max}) based on generated DVHs was done.

With heart and lung contoured as OARs, QUANTEC and RTOG constraints were applied. The average cardiac doses were 21.82Gy and 16.8Gy for TwF and FiF respectively. Analysis of generated DVHs based on the two techniques suggested superior dose coverage of V95 = 48.91 for the FiF compared with V95=46.20 for the TwF. The FiF approach generated superior dose homogeneity indices compared with the TwF treatment planning approach

Keywords: conformal, segmented, wedged-field

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The author expresses sincere gratitude to the oncology team and the research department of the Sweden Ghana Medical Centre for the diverse efforts towards this study.

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PS4 (T4.4-0905)

Nicaragua: The experience of five years of digital equipment and Quality Controls

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Performing quality controls on Ray X machines in Nicaragua has been a task for the Laboratory of Radiation Physics and Metrology (LAF-RAM) of National Autonomous University of Nicaragua. The laboratory has more than 20 years of experience on radiation protection and metrology this is why every year the LAF-RAM is authorized by national authority body to carry out not intrusive inspections and quality controls to equipment such as conventional radiology, mammography, tomography, dental and interventional radiology equipment.

In recent years, there has been noticed an increment on moving towards digital technology in most of the evaluated X ray equipment and medical institutions. Since 2014 the Laboratory has carried out 200 quality controls and it was found that there is still a significant presence of film-screen technology specially on Conventional radiology equipment with 66% CR type, 26% were film-screen and 7% DR type, including portable X Ray machines. In the case of mammography 61% of the evaluated equipment are CR, 25% film-screen 14% are DR type. For intrinsic digital equipment such as CT it was found that only 30% of them complies all the requirements for image quality and dosimetry.

Transition to digital quality control began by adapting the protocols from film-screen to digital technology. However, the main obstacles faced so far, have been the acquisition of proper phantoms for assessment of physical image quality, but more important is to deal with the disadvantage of not been able of getting images on raw format due to the common practice of manufacturers of deactivate this feature at the moment of acceptance testing on medical institutions.

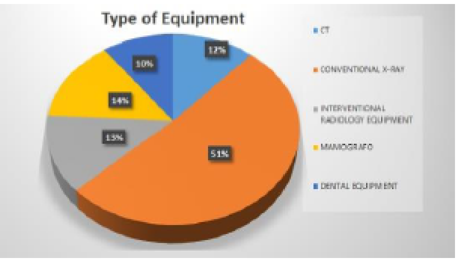

Fig. 1. Percentage of QC evaluated by modality from 2014 to 2019

Keywords: *Quality Control, DR, CR*

Poster Session T4

PS4 (T4.4-0922)

Image acquisition technique for nuclear medicine using deep profile learning

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Purpose: There are several main methods for image reconstruction technique for nuclear medicine. (ex. FBP, MLEM, OSEM, etc.). In order to overcome disadvantages of conventional technique such as noise, weaken signal, we proposed the image acquisition technique using deep profile learning.

Methods: Nuclear medicine image of 120 cases have been acquired by using Monte Carlo simulation. Over 10,000 profiles have been measured from all images to train the image acquisition system. And we prepared the profiles of ideal pattern from ground truth image regarding each reconstructed image. After the training with the profiles (X-label) from reconstructed image and ideal profile (Y-label) through the deep neural network using softmax, we inserted sinogram of new cases to image acquisition system to get image.

Results: The deep neural network exported optimized profile from new sinogram. We re-arranged these optimized profiles according to original order. The re-ordered profiles matrix can show the optimized image which has better image quality than the performance of conventional image reconstruction technique.

Conclusions: This study shows a feasibility about the image acquisition using the deep profile learning for nuclear medicine imaging with good performance.

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Acknowledgement: Add acknowledgement here if required, for example, the funding body of the research work.

Keywords: Nuclear Medicine, Monte Carlo simulation, Deep learning, Image reconstruction

PS4 (T4.4-0926)

Development of Infrared (IR) Marker for Thermoplastic Immobilization Tool

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Introduction: In radiation therapy, the thermoplastic immobilization tool is one of the most commonly used fixing methods in clinical practice. However, In the case of the thermoplastic immobilization tool, it is difficult to use the patient motion monitoring method using the camera equipment in the radiation treatment because of the nature of the fixing method. To overcome these drawbacks, we propose an Infrared (IR) marker designed to enable camera-based patient movement monitoring while using a thermoplastic fixation device.

Materials and Methods: Modeling of the IR marker was performed using 3D modeling software. The structure of the IR marker was consisting of a spherical IR marker, a cylinder to be attached to the skin and a disk-shaped structure between the IR marker and cylinder. The disk-shaped structure has a number of holes to be combined with the cylinder. This design of the disk is to free position of the cylinder so that there is no interference when placing the IR marker on the surface of the body. The modeled IR marker is output to the 3D printer and placed on the thermoplastic immobilization tool to verify that there are no problems.

Results: IR marker for thermoplastic was developed to detect patient surface movement in the thermoplastic immobilization tool. The IR marker was printed using a 3D printer. When the IR markers were placed on the thermoplastic immobilization tool, we confirmed that it is well located on the thermoplastic immobilization tool without any other problems. In addition, although it is not a quantitative evaluation, we have confirmed that the IR marker moves according to the motion of the patient.

Conclusion: We have developed the IR marker for the thermoplastic immobilization tool and the developed IR marker was confirmed to be suitable for our purpose. Further study, we will verify with the stereo vision that the developed IR marker detects movement of the patient's surface.

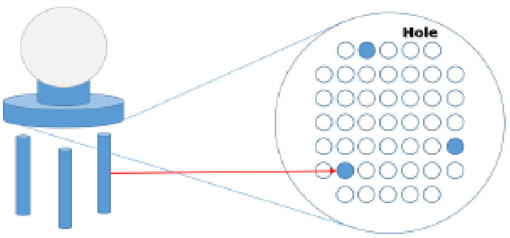

Fig. 1. The modeling of IR marker for thermoplastic immobilization tool

Keywords: Infrared marker, Thermoplastic, 3D printing

ACKNOWLEDGMENTS
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PS4 (T4.4-0927)

Sensing changes in tumor during boron neutron capture therapy using PET with a collimator: Simulation study

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The purpose of this study was to demonstrate the feasibility of sensing changes in a tumor during boron neutron capture therapy (BNCT) using a Monte Carlo simulation tool. In the simulation, an epi-thermal neutron source and a water phantom including boron uptake regions (BURs) were simulated. Moreover, this simulation also included a detector for positron emission tomography (PET) scanning and an adaptively-designed collimator (ADC) for PET. After the PET scanning of the water phantom, including the 511 keV source in the BUR, the ADC was positioned in the PET's gantry. Single prompt gamma rays were collected through the ADC into a neutron beam. Then, single prompt gamma ray-based tomography images were acquired of variably sized tumors by a four-step process. Both the signal-to-noise ratio (SNR) and tumor size were analyzed from each step image. From this analysis, we identified a decreasing trend for both SNR and signal intensity as tumor size decreased which was confirmed in all images. In conclusion, we confirmed the feasibility of sensing changes in a tumor during BNCT using PET and an ADC through Monte Carlo simulation.

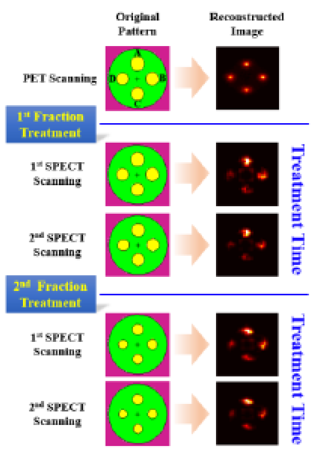

Fig. 1. A diagram of the original patterns and reconstructed images depending on the BNCT procedure

Keywords: *Boron neutron capture therapy, PET, Adaptively-designed collimator*

ACKNOWLEDGMENTS

This study was supported by the Development of Radiation Medicine Technology (Grant No. 2017M2A2A7A01021264) through the National Research Foundation of Korea funded by the Ministry of Science, ICT & Future Planning, Republic of Korea.

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PS4 (T4.4-1015)

Release Criteria of Treated Animals with Radionuclides

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In this study, release criteria of animals treated with radionuclides were introduced. The criteria were considered for licensed radionuclides for the purpose of veterinary medicine uses. For the practical use in institutes, the criteria were derived as external radiation level (dose rate) based on an effective dose 1 mSv to an individual. For unrestricted use of the released animals, conservative assumptions were introduced. For example, radioactive materials in animals were assumed as point source (ignored distribution of radioactive material and no shielding by tissue) and only physical half-lives of radionuclides were considered. Also the distance between person and animal was assumed very close (15cm or 30cm) with 1/3 occupancy factor (8 hours a day). The nominal values were listed in table 1. Under the domestic regulation, animals treated with radionuclides were considered as radioactive. Therefore, the licensee, who use radionuclides for veterinary medicine purpose, should make every effort to maintain dose to other individuals as low as reasonably achievable, such as assessment of individual dose from released animals and distribution of written instructions. In applying the criteria for release of animals, patient-specific information such as actual pattern with animal keepers or owners, should be considered.

Table 1. Dose rates for release of animals treated with radionuclide

Radionuclides	Dose rate (mSv/hr) at 15cm
Cu-64	0.04
I-124	0.005
I-131	0.0025
In-111	0.0077
Lu-177	0.003
Sn-117m	0.0015
Tc-99m	0.0865
Zr-89	0.0066

Keywords: Release criteria, animals treated with radionuclides, veterinary medicine

ACKNOWLEDGMENTS

This work was supported by the Nuclear Safety Research Program through the Korea Foundation of Nuclear Safety (KoFONS), granted financial resource from the Nuclear Safety and Security Commission (NSSC), Republic of Korea. (No. 1803013-0118-CG110)

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PS4 (T4.4-1031)

Novel electron arc technique for re-irradiation of wide-spread skin cancer on the extremities

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Kaposi's sarcoma (KS) is a radiation-responsive skin cancer typically occurs on the upper or the lower extremities. Megavoltage photon beam with two opposite technique or volumetric modulated radiation therapy (VMAT) are used to treat this disease. However, it is challenging to achieve enough dose saving of underlying normal organs with the conventional photon beam techniques, especially for recurrent KS. For the purpose of treating the recurrent patient by minimizing radiation exposure to the normal organs a novel electron arc treatment with a scatterer is proposed in this study for wide spread skin cancer on the lower extremities

Monte Carlo simulation was used for dose calculation. Electron beam parameters for MC model: the incident electron parameters in terms of mean energy, energy spread, position spread and angular spread were accurately tuned to achieve < 3%-agreement with measurements. After tuning the MC model, a feasibility study of electron arc technique was performed using the MC simulation and a dosimetric study with a phantom to validate the dose distribution on the skin with the electron energy, the field size, and spoiler usage as variables. Plan comparison among the electron arc therapy, the opposite photon beams, and VMAT was performed for a recurrent KS on the lower extremities who was treated using opposite photon beams.

The scatterer around the treated extremities was used to increase scattered electrons and thus improve uniformity of electron fluence around the target area. Penetration depth of electron beams was 9.1 and 13.9 mm and full width at 80% of maximum dose in longitudinal direction was 20.7 and 23.2 cm for 4 and 6 MeV electron arc beams, respectively. Inside the scatterer, maximum absorbed dose was occurred at the center of the scatterer and it was continuously decreased up to 70% of maximum dose as close to the inner surface of the scatterer.

Dose homogeneity (D5% /D95%) of PTV was 1.93 for 4 MeV and 1.44 for 6 MeV electron arc beams, while 1.09 for two opposite photon beams and 1.06 for VMAT. Mean doses to soft tissue and bones were as low as 23.5% and 3.4% of the prescription dose in 4 MeV electron arc plan and 51.0% and 26.0% in 6 MeV electron arc plan, while 99.2% and 98.8% in two opposite photon plan and 73.1% and 69.9% with VMAT plan.

In conclusion, since the electron arc therapy provides much better dose saving to the normal organs, this technique could be an option for recurrent skin cancer treatment on extremities.

Keywords: *Re-irradiation, Skin cancer, Electron radiation therapy*

ACKNOWLEDGMENTS

This work was supported by a grant (2019-7049) from the Asan Institute for Life Sciences, Asan Medical Center, Seoul, Korea.

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PS4 (T4.4-1117)

Qualitative Improvement of Effective Dose Area Product for Patient Dose Area Product on Medical Radiation Monitoring System

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The patient radiation dose of general radiography and fluorography were recommended to Dose Area Product (DAP). However, the usage of DAP raises concerns about overestimating the actual dose of radiation exposure. The purpose of this study was to introduce the fundamental of Effective Dose Area Product and develop an EDAP extraction algorithm using Digital Imaging and Communications in Medicine (DICOM). EDAP was to more accurate than DAP in patient radiation dose. The EDAP was calculated by DICOM tags information, such as DAP, air kerma, and collimated DICOM tag information. The DICOM tag number (0018,115E) showed DAP value of the patient. This value was calculated by each X-ray exposure condition or measured dose area product meter of exposure. DAP value was calculated by multiplying the area of exposure by air kerma at reference point. The value includes real patient area and blank area in exposure area. Developed gateway was divided blank area and real patient area using image processing. And then this gateway recalculates DAP of real patient area multiplied by air kerma. We developed a system was extracted to related DICOM information and real patient area of image processing algorithms and then calculated to EDAP value. Finally, the system created a new DICOM tag and sent to dose monitoring system. EDAP value will be useful information on patient radiation quality control and dose reduction program for general radiographic department.

Dose area product = exposure area × air kerma at reference point

Effective dose area product = (exposure area – blank area) × air kerma at reference point

Figure 1 showed EDAP gateway screen shot. The original DAP value was 138.9382 mGy·cm^2 but EDAP value was 106.982 mGy·cm^2. The EDAP value was 23% smaller than original DAP value.

Fig. 1. This figure showed effective dose area product (EDAP) gateway screen shot.

Keywords: Dose Area Product, Effective Dose Area Product, DICOM

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PS4 (T4.4-1203)

Radiation Dose Due to PET and PET-CT

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Nuclear medicine is a branch of medicine that uses radiopharmaceuticals to diagnose and treat metabolic abnormalities in the body. Frequency of nuclear medicine has been increased constantly. Increasing frequency of nuclear medicine may increase radiation dose to the public. PET and PET-CT are major examinations of nuclear medicine. For the management of radiation dose due to nuclear medicine, it is necessary to assess radiation dose due to PET and PET-CT. The purpose of this study was to assess the collective dose and effective dose per capita due to PET and PET-CT in Korea. In order to assess radiation dose due to PET and PET-CT, we collected frequency data of PET and PET-CT through the Health Insurance Review and Assessment Service (HIRA). We developed classification system for PET and PET-CT using the collected frequency data. PET and PET-CT examinations were classified into nervous system, oncology, cardiovascular system, and musculoskeletal system. We analyzed the collected frequency data using SAS program. The frequency of PET and PET-CT was approximately 0.19 million. Oncology examination was the highest with 0.17 million (about 92%). In order to assess the effective dose per examination, we investigated the activity of radiopharmaceuticals. Dose conversion factors given in International Commission on Radiological Protection (ICRP) were used for the dose calculation [1, 2]. Activity of radiopharmaceuticals was given in Nuclear Safety and Security Commission (NSSC). If radiopharmaceutical activity data was insufficient, foreign literatures were referenced [3, 4]. Collective dose of PET and PET-CT was approximately 730 man·Sv. Oncology examination was the highest with 590 man·Sv (about 81%). Effective dose per capita of PET and PET-CT was approximately 13.98 µSv. Oncology examination was the highest with 11.44 µSv (about 81%). The results of this study can be used to manage radiation dose by PET and PET-CT.

Table 1. Frequency and radiation dose of PET and PET-CT in Korea

Examination	Frequency	Collective dose (man·Sv)	Annual effective dose per capita (µSv)
Nervous system	13,905	129.30	2.52
Oncology	172,770	587.83	11.44
Cardiovascular system	184	0.63	0.01
Musculoskeletal system	88	0.53	0.01
Total	186,947	728.28	13.98

Keywords: PET and PET-CT, Nuclear medicine, Medical radiography, Public dose

ACKNOWLEDGMENTS
This work was supported through the KoFONS using the financial resource granted by NSSC. (No. 1803013)

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PS4 (T4.4-1221)

Standardization of cardiac amyloid measurement using Tc-99m PYP

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Cardiac amyloidosis is one of the most expeditiously progressive forms of heart disease, characterized by extracellular deposition of an indissoluble protein-amyloid in the myocardium. Median survival from diagnosis is <6 months for light chain amyloidosis to 3-5 years for transthyretin amyloidosis. A long asymptomatic period and an unfavorable prognosis for decompensation of heart failure make the diagnosis of cardiac amyloidosis at the early stages of the disease extremely important. Currently, radionuclide imaging is one of the leading positions in the diagnosis of amyloidosis of the heart. Early studies have shown high accuracy of diagnosis cardiac imaging with bone avid radiotracer ^{99m}Tc-PYP for the diagnosis of cardiac amyloidosis (sensitivity and specificity> 90%). When the effectiveness of ^{99m}Tc-PYP is proven, the question of standardizing the diagnostic protocol remains open. In our work, we tried to answer questions about the correct time for diagnostics and determine the mechanism of binding of ^{99m}Tc-PYP. Purpose: Determine the difference of mechanism of ^{99m}Tc PYP accumulation to the heart and to the bones, and standardize the diagnosis procedure for cardiac amyloidosis. Methods: 28 patients of Nagasaki University Hospital underwent ^{99m}Tc-PYP planar and single-photon positive emission computed tomography cardiac imaging. Cardiac retention was evaluative with both a semiquantitative visual score (range 0 - no uptake to 3 - diffuse uptake) and by quantitative analysis by drawing a region of interest (ROI) over the heart corrected for contralateral counts and calculating a heart-to-contralateral ratio (H/CL). H/CL was calculated with 1hour images and 3hour image, then compared with ANOVA. Results: Among patients who underwent ^{99m}Tc-PYP planar and SPECT cardiac imaging, H/CL ratio was significantly higher in 1hour as compared with 3hour acquisition (1.79 vs 1.60; $p<0.001$). Conclusion: The diagnostic accuracy of ^{99m}Tc-PYP after 1hour injection had superior efficiency whereas imaging at 3hour. For the standardization and proper diagnosis, measuring H/CL on 1hour should be more suitable than that on 3hour.

***Keywords**: cardiac amyloidosis, 99^{m} Tc-PYP, nuclear imaging.*

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PS4 (T4.A-0199)

Ethical and Other Considerations in Conveying Information on the Benefits and Risks of Diagnostic Medical Radiation Procedures to Patients

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This presentation will consider the ethical value of Autonomy as a human right enshrined in international conventions and as applied to diagnostic medical procedures involving exposure to ionising radiation.

In medical procedures, including those involving exposure to ionising radiation, there are always associated risks to be weighed against the perceived benefit, but also by comparison there are the benefits and risks of alternative procedures or modalities, or of not doing the procedure at all.

Patient autonomy must be gained through Informed Consent - through the patient being given appropriate information about all of the perceived benefits and risks of undergoing the procedure, as well as the alternatives, so that they can make a choice as to whether the procedure is carried out.

The presentation will consider how such information on benefits and risks may be weighed and presented to the patient.

Keywords: Ethics, Informed Consent, Medical Exposures

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PS4 (T4.A-0649)

Patient safety (both mother and child) as strategic priority – the ethical aspects of optimization of RT treatment plan for pregnant patient with Hodgkin lymphoma

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The radiotherapy of a pregnant patient is a sensitive topic, and therefore the patient safety (both mother and child) should be considered as strategic priority. The optimization in radiotherapy means the delivery of prescribed dose to the target volume while keeping the dose to the organs at risk (OAR) at a level as low as possible. This is especially important for young patients and those with predicted long survival time after treatment, which concern most of the patient diagnosed with Hodgkin Lymphoma. During such a optimization for pregnant patient not only the risk for the fetus should be taken into account but also the risk for the mother. This requires possible reduction of dose to all organs including heart, lung, breasts, salivary glands, bone marrow, spinal cord, esophagus. The dose to the fetus can be significant reduced by minimizing the volume irradiated tissues of the mother (if possible and clinically acceptable).

One of the change over the last years is that the ISRT (Involved Site Radiation Therapy) has replaced the MANTEL fields approach, which has contributed to lower long term toxicity of the radiation treatment. In some countries the MANTEL fields technique might still be preferred due to various reasons such as the fact that this treatment is quite easy to be prepared by medical staff (RTT, MD, MPE) or due to local condition like the type of insurance, which does not allow to propose complex treatment. This approach contradicts the ethics principle, as the ISRT conformal treatment maintains benefits of therapy while reduces toxicity to the patient, therefore no matter what is the insurance policy the ISRT should be performed. If the local technical capacity does not allow to propose the ISRT instead of MANTEL fields, according to the ethics principle, the medical staff should at least inform the patient where it is possible to try to undergo much less toxic treatment. In order to explain this issue a comparison of OAR's doses and radiation fields might be shown to the patient. An example of such comparison is shown in Table 1 and in Fig. 1.

Table 1. Comparison of OAR's doses (RT scheme: 30 Gy in 15 fractions)

OAR	MANTEL fields	ISRT VMAT
Heart	V25 = 85,5 %, V30 =71,9%, Dmean= 27,28 Gy	V25 = 71,9 %, V30 =8,1%, Dmean= 11,71 Gy
Lungs (L+R)	V20 = 51,9 %, V30 =36,6%, Dmean= 18,34 Gy	V20 = 10,1 %, V30 =1,6%, Dmean= 9,60 Gy
Salivary gland	Dmean(L)= 31,38 Gy, Dmean(R)= 31,14 Gy	Dmean(L)= 5,00 Gy, Dmean(R)= 4,63 Gy
Left Breast	Dmax=31,29 Gy, Dmean= 6,00 Gy	Dmax=18,27 Gy, Dmean= 3,71 Gy
Right Breast	Dmax=30,25 Gy, Dmean= 5,31 Gy	Dmax=14,88 Gy, Dmean= 2,70 Gy

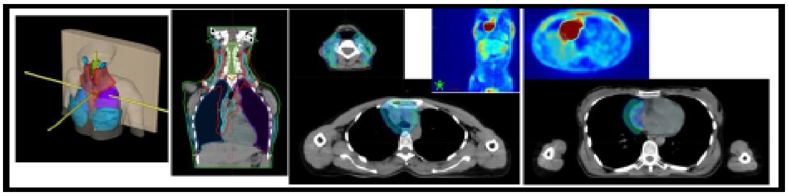

Fig. 1. PTV- ISRT based on PET/CT examination performed before treatment & 95% izodose.

***Keywords**: Hodgkin Lymphoma in Pregnancy, Radiotheraphy during pregnancy, ethics*

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PS4 (T4.B-0277)

Assessment of Radiation Exposure of Healthcare Workers from Patients Receiving Nuclear Medicine Procedures

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Especially since the Fukushima nuclear accident in 2011, patients, families, and hospital staff have been increasingly concerned about radiation exposure and the potential for adverse effects from diagnostic and therapeutic uses of radiation. The field of nuclear medicine uses radioactive materials for both imaging and therapy, including the combined utilization now called theranostics. The linear no-threshold (LNT) model is used for risk prediction as well as to guide regulatory limits and radiation protection procedures. However, it seems evident that the LNT model substantially overpredicts the concerns for very low doses of radiation (much below 100 mGy), and may in fact ignore beneficial aspects (hormesis) in this low dose range. Nuclear medicine and other diagnostic imaging modalities contribute radiation doses to patients and hospital staff that are well within this low-dose range. Here we report a study of the levels of radiation dose received by hospital staff due to routine use of nuclear medicine procedures. The goal is to increase the knowledge of hospital staff regarding nuclear medicine procedures and the very low levels of exposure received from these.

Thermoluminscent dosimeters (TLDs) were worn by hospital staff in various medical specialty wards of 14 hospital units, representing a range of facilities that conduct large numbers of nuclear medicine procedures, such as whole body bone scans and first-pass ejection fraction cardiac imaging using Tc-99m radiopharmaceuticals. Staff in these wards are potentially exposes by patients who have received inpatient imaging during the course of their hospital stay. Some TLDs were worn continuously (24 hours per day) by multiple workers while others were worn only during the regular shift. TLDs were also placed in a variety of fixed locations in these units, such as on door frames and above nurses' stations, where hospital staff would pass frequently or spend substantial periods of time. All TLDs were in place over a period of 40 days. TLDs were read by an independent certified agency. The TLD readings from both the staff dosimeters and the fixed dosimeters showed only background levels of radiation dose, comparable to a control TLD that was placed in a location away from the wards. These results showed that the radiation exposure to staff in these hospital units was not elevated compared to background radiation levels. The radiation exposure to hospital staff and public due to nuclear medicine procedures was minimal, and concern about excessive radiation exposure from these procedures is unwarranted.

Table 1. TLD readings at fixed locations

Specialty ward	Location	Dose equivalent (μSv)
Genitourinary	Changing room	Background
	Nurses station	Background
	Entrance	Background
Emergency	Observation area - 1	Background
	Observation area - 2	Background
	Entrance	Background
	Observation area - W	Background
	Nurses station	Background
	Diagnosis room	Background

Keywords: Nuclear medicine; Radiation dose; LNT model

PS4 (T4.B-0718)

Contribution of the LAPRAM Network to improve Radiological Protection in Medicine in Latin America

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In order to verify the progress in the implementation of the Bonn Call to Action, identify problems and possible solutions, as well as define indicators of progress in these actions, the Ibero-American Conference on Radiological Protection in Medicine (CIPRAM) was held in Madrid, Spain in October with the support of national and international organizations.
In August 2017, the International Symposium on Radiation Protection in Medicine was held in the city of Arequipa, Peru, where there were 221 participants from 16 countries with the aim of discussing and identifying the main problems presented and proposing solutions regarding to the Call of Bonn to Action and the CIPRAM agreements with the active participation of professionals from health institutions, regulatory bodies, Health authorities, Professional Societies, Radiation Protection Societies and Medical Physics , from universities and private organizations in the Latin American region.
The Symposium represented the opportunity for the exchange of experiences and the proposal proposal, raising the need to disseminate in all the sectors involved, the proposals of the mentioned events, show the good experiences, propose solutions to the problems that exist in the Latin American region to in order to improve the conditions of radiation protection but also to follow up the agreements with the participation of professionals and organizations in the region.
For this reason, at the Arequipa Symposium it is decided to create a working group to comply with the aforementioned agreements. At the end of 2017, the group takes the name of the Latin American Network for Radiological Protection in Medicine (LAPRAM Network) composed of specialists from the region who voluntarily collaborate by carrying out various activities with the general objective of strengthening radiation protection in medical applications of radiation Ionizers in the region.
Some of the actions that the LAPRAM Network has carried out so far are:
- Incorporation of specialists from the region
- Creation of the profile on Facebook: www.facebook.com/redlapram with free access and with 6951 followers
- Translation of IAEA and WHO technical documents
- 15 Webinars
- Formation of working groups to address specific issues
- Dissemination of the activities of the Network in various national and international events
- Event sponsorship

The creation of the LAPRAM Network has been well received at the XI Regional Congress on Radiological and Nuclear Safety held in Havana, Cuba, from April 16 to 20, 2018 and is mentioned in the agreements of the Congress.
The LAPRAM Network supports the organization of the II International Symposium on Radiological Protection in Medicine to be held in Guayaquil, Ecuador in November 2020.

PS4 (T4.B-0757)

On Empathy in Radiation Protection

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Nuclear issues are known to evoke strong emotional reactions. Radiation protection experts translate and communicate scientific knowledge derived from (medical) research and follow-up data from nuclear events (ICRP103) to members of the public such as patients undergoing radiotherapy treatment, cleaning staff in the hospital working on the nuclear medicine ward, members of the hospital emergency team and to people who live within a twenty kilometer radius of a nuclear power plant. Since the theme of the conference is – Widening public empathy towards radiation protection – let's have a closer look at empathy.

On empathy

Empathy is the ability to attribute mental states to one self and others and consists of at least two components, a cognitive (understanding) component and an emotional, affective (feeling) component which entails an appropriate affective response to the other person's mental state (Bellet, 1991; Baron-Cohen, 2006). When assessing empathy score in an average population the results show that the majority of people are empathic (Baron Cohen, 2011). Physician empathy towards patients has shown to be a positive treatment factor since patient safety in radiotherapy treatment is increased through patient education and involvement (Bibault, 2016). Another study among physician specialists regarding empathy towards their patients with cancer shows that an empathic physician has a beneficial outcome to patient overall wellbeing. Physicians, however, also stated that feeling into a patient can take its emotional toll by emotional contagion (Robieux, 2018). So, where does empathy fit into radiation protection?

Human experiences in nuclear issues and technology regarding empathy

Radiation protection officers mostly embark on explaining radiation risks and protection principles based on scientific evidence on a cognitive level. In order to empathize on an emotional, affective level more insight in human experiences in nuclear issues is needed. Michael Edwards (2018), a psychiatrist-researcher, looked into human experiences regarding nuclear energy and technology. One of the many findings in this elegant qualitative study is that the fear of radiation appears to involve multiple other, often unconscious fears, some of which are activated because of the general state of anxiety and not necessarily related to the actual radiation threat. Helping those affected manage anxiety overall is seen as just as important as helping them deal with radiation concerns. How do we take this information into account as radiation protection experts?

Empathy in radiation protection

Radiation protection experts need to use expert scientific knowledge as well as empathic skills when communicating radiation risks and protection with members of the public. The experts need to be aware and empathize with possible underlying unconscious fears at play when communicating and explaining radiation risks and protection. As shown before most people are empathic and skills regarding empathic communication can be expanded by practical training. Furthermore, empathy should not be confused with trust, which entails the confidence in the honesty and integrity of a person or thing. Ultimately, it is trust that matters, and empathy is an essential prerequisite to gain trust. Therefore, when communicating radiation risks and protection the ultimate goal should not be widening public empathy but to widen public trust. Radiation protection experts need to communicate scientifically-sound information in an empathic manner in order to gain and/or increase trust of members of the public towards radiation protection.

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PS4 (T4.B-0956)

Radiological risk perception in medicine: current issues and challenges found in a national survey

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Radiological risk perception is the result of a complex combination of knowledge, cultural background, emotions of relative risks, and messages and information received from trusted persons and organizations. In order to know about this subject, a national survey was organized in San Carlos de Bariloche, the biggest city in Argentine north-western Patagonia. In addition to its beautiful and dreamy landscapes, this location is famous worldwide because of the scientific activities carried out there, most of them related to nuclear and radiation technologies. The city has expanded around an atomic center, where a research reactor has operated for the last 37 years, and an internationally well-known nuclear company and several hospitals and health centers have settled down nearby. However, despite its radiation-related history, there are no local records about radiological risk perception; even at national levels little information can be found.

A fifteen-question dedicated questionnaire was designed including socio-demographic variables (such as gender, age and profession) and questions about work environment, risk perception and risk communication were asked. A five-point Likert-type scale (from strongly agree, agree, neutral, disagree, to strongly disagree) was used for most of the questions. Also, surveyed people were asked to compare radiological risks in medicine with other daily and familiar risks such as those related to smoking, sports and leisure activities. A printed version and a web-based version were distributed among people personally, by e-mail and through social networks. The survey was voluntary and anonymous. The participants pool consisted of 223 individuals (137 women and 86 men, age: 18 to 72 years old) from all over the country and 10 Argentinian radiation protection experts. The data were analyzed using the Epi Info™ software package version 7. The descriptive statistics (such as mean and percentage) were used to present distribution of the socio-demographics and the respondents' risk perceptions.

In brief, results showed good agreement between risk perception of laypeople and experts; however, a negative correlation could be seen between these two groups when they were asked about the cancer incidence when living near a nuclear power plant. Regarding risk communication, respondents said they did not trust in short messages found in social networks. Finally, a positive correlation was found between the perception respondents had about health risks and the most important causes of death in the country. An important number of surveyed individuals replied that undergoing CT exams was a high-risk activity for an individual. Respondents believe the information given to the patients undergoing radiation exposures in medicine is reliable and enough and they think health professionals are well-trained in radiological protection.

Although there is no right or wrong risk perception, a lot of work must be done in order to build an agreement between radiological risk perceptions and risk assessment due to radiation medical exposures. It should involve working on the design of public policies focused on radiological protection training for health staff, strengthening communication skills and channels, and most importantly, the recognition of the radiological protection of patients as a public health issue.

Keywords: radiological protection, risk perception, Argentina

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PS4 (T4.C-0477)

The Rising Generation: SRP's work with Young Professionals

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In 2010, the Society for Radiological Protection (SRP) launched its "Rising Generations Group" (RGG) for those people within the first ten years of their career in Radiation Protection.

Over the time it has been active, the group has been instrumental in developing many initiatives to promote the inexperienced generation of professionals both within and outside of the society. This has included the annual RGG presentation prize and the biennial UK Young Professional Award (which acts as the UK's selection process for the IRPA YGN competitions at the European meetings and IPRA Congresses), both of which take place at the SRP's Annual General Meeting and Conference.

The latest initiative has been the introduction of a mentoring scheme, which matches professionals with more experienced individuals to promote scientific, technical, or professional mentoring. This scheme has had an unexpectedly positive response and a very high uptake.

In the lead-up to its 10th anniversary year, the group has looked back over its history, its accomplishments, and its close ties with the IRPA Young Generation Network. More importantly, it has looked forward to its future and to the future of professionals in the first ten years of their career in the world of radiation protection.

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Acknowledgments go to the SRP, without whom the RGG would not exist and all of this work would not happen

PS4 (T4.C-0954)

Evaluation of dose enhancement of x-rays in thyroid tissue labeled with gadolinium and gold nanoparticles using MCNPX code

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Conventional radiotherapy, which uses photon beams, is the most common method used to treat local tumors.This method will destroy the cancerous tissues on the one hand and protect the healthy tissues on the other.In the early years, the study of photon therapy using nanoparticles that have accumulated within cancer tissues has opened a new chapter of research.In the present study, to evaluate the dose improvement in thyroid tissue, the photon activation method was used by labeling the thyroid tissue with heavy activating elements such as gadolinium and gold and X-ray irradiation.For this purpose, the phantom of normal thyroid tissue activated by nanoparticles was modeled using MCNPX code.The phantom consists of a cube of 16 cm with water and a tumor of 2.2 cm with soft tissue.Nanoparticles were inserted into the tumor at concentrations of 25, 10, 50 and 75 and at radius of 15, 45 and 60 nm to determine the dose improvement and the ratio of increasing dose in the presence and absence of nanoparticles was investigated.[1]

Conclusion: The results of executing the program was observed that increasing the nanoparticle radius increased the absorbed dose and increasing the concentration of nanoparticles also increases the absorbed dose in tumor tissue.For all concentrations, gadolinium performed better than gold so that the best coefficient of improvement of the gadolinium dose to the nanoparticles radius was 60 nm with a value of 1.71.At low concentrations, gold absorption dose changes were more sensitive to the nanoparticle radius but with increasing concentration, gadolinium becomes more sensitive to radius changes.

Discussion: The nanoparticles increase the absorbed dose in the cancerous tissue and decrease the dose in the surrounding tissue.Also, the absorbed dose in the thyroid tissue labeled with activating elements is significantly increased compared to normal thyroid tissue.

Keywords: *Thyroid cancer, Gadolinium and gold nanoparticles, MCNPX code*

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PS4 (T4.C-1107)

Importance of Ensuring Safety in Radiation Medicine as Part of imPACT Reviews and National Cancer Control Planning

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Nuclear technology plays a key role in the diagnosis and treatment of cancer. Over the past six decades the IAEA has strong technical expertise and experience in the delivery of nuclear medicine and radiotherapy technology.

The IAEA provides assistance to Member States through the education and training, technical advisory services, the provision of materials, tools and equipment, and the support of radiation safety and quality assurance.

PACT provides assistance in the area of cancer through imPACT Reviews, resource mobilization and by supporting the development of strategic documents, such as Comprehensive National Cancer Control Plans (NCCPs) and bankable documents for fundraising. It also supports cancer-related IAEA activities that are delivered through technical cooperation, human health and another IAEA programmes.

The imPACT Reviews assess a country's cancer control capacities and needs and identify priority interventions to effectively respond to its cancer burden. Using this service, we can better understand the overall situation in cancer control in a Member State including actions related to radiation protection, which would be reflected in the Comprehensive National Cancer Control Plan.

Ensure that radiation medicine is used safely and securely and in a most beneficial way without harm to the patients, healthcare workers or the public is an important element considered in imPACT reviews and in comprehensive cancer control planning. PACT and the Division of Radiation, Transport and Waste Safety (NSRW) collaborate through the imPACT reviews to ensure that this is done in assessments and plans.

Keywords: *Comprehensive National Cancer Control Plans (NCCPs), IAEA, imPACT Reviews, Radiation medicine, Radiation Protection, Safety*

PS4 (T4.D-0023)

A holistic approach for risk analysis in therapeutic nuclear medicine in Cuba

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The most useful prospective methods to risk analysis in medical practices with ionizing radiation are the risk matrix (RM) and the failure mode and effect analysis (FMEA). In another hand, at the world level developed various systems for reactive risk analysis (ILS), for example ARIR, ROSIS and SAFRON. In the state of the art, RM, FMEA and ILS are not matching. This study aims to identify the most contributors to the radiological risk for radionuclide therapy (RT) applying a holistic approach useful for decision makers and as a training tool for staff for strengthen safety culture.

Developed generic models for RT were adapted to six cases in nuclear medicine services in Cuba. This includes the radiosynoviorthesis and the myelosupressor treatment with Phosphorous 32 of polycythemia Vera. The TG-100 of AAPM was taken as reference for RT. For safety assessment are used a new combined methodology and a Cuban code SECURE-MR-FMEA version 3.0, which increases the efficacy and efficiency in this study.

The application of generic models shows a selection of 63% of the total accidental sequences, 76% of barriers, 58% of frequency reducers and 50% of consequence reducers, as minimum. For patient specific treatment, these were higher than 91%. Most important identified steps of work processes, control elements and root causes for the risk showed as integrators of the improvement quality and safety plan.

In addition, there are an informational compendium made with Dreamweaver version 8.0 and an international incident database (IDB) with around 30 years of published events, which includes near misses for this practice, and a standard list of root causes and adapted severity scale. This research allows identifying priority measures to keep exposure optimization for patients, workers, and public. The developed tools could applied to the rest of medical practices and for continuous learning in our organizations.

Keywords: *risk matrix, radiological risk, radionuclide therapy.*

PS4 (T4.D-0097)

Clinical Indication Based (CIB) Diagnostic Reference Levels (DRLs) for Contrast Radiography Examinations: A Guide for Radiation Safety culture and good clinical practice

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Introduction: Indication specific DRLs is pivotal in optimization of medical procedures and it serves as a guide to Radiology practitioners in achieving the international recommendations and current trends by International Commission on Radiological Protection (ICRP).

Objectives: To establish CIB DRLs for contrast radiography examinations in two teaching hospitals in North Eastern Nigeria.

Materials and Methods: A Prospective cross - sectional study was conducted in two major University Teaching Hospitals. Three hundred and Sixty (360) patients participated in the study. Doses were recorded using thermo-luminescent dosimeter (TLD) chips and dose area product (DAP) meter. Student T-test was used to determine the relationship between the mean entrance skin doses (ESD) obtained in the two centers while Pearson's correlation was used to determine the relationship between the dose and anthropo-technical parameters. Statistical significance was set at $p < 0.05$.

Results: Findings showed that the clinical DRLs for this study were 6.68 mGy and 10.66 mGy.cm^2 (IVU), 2.31 mGy and 3.67 mGy.cm^2 (HSG), 2.66 mGy and 8.98 mGy.cm^2 (barium meal), 12.78 mGy and 20.64 mGy.cm^2 (barium enema), 2.73 mGy and 6.56 mGy.cm^2 (barium swallow), and 2.05 mGy and 7.77 mGy.cm^2 Retrograte Urethrography (RUG), respectively. The Entrance Skin Dose (ESD) and Dose Area Product (DAP) showed statistically significant relationship with technical parameters ($p < 0.05$) for barium enema. The remaining studies showed no statistical significance ($p > 0.05$).

Conclusion: Clinical DRLs in this work recorded lower values. However, regular dose optimization technique and etiquettes are required to ensure good practice.

Key words: Barium Meal, Barium Enema, Barium Swallow, Hysterosalpingography (HSG), Intravenous Urography (IVU)

PS4 (T4.D-0372)

A framework for enhancing radiation safety culture in health care

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The term "safety culture" was first defined by the IAEA-INSAG as the assembly of characteristics and attitudes in organizations and individuals which establishes that, as an overriding priority, nuclear plant safety issues receive the attention warranted by their significance (1). This concept originally was further developed and expanded to other industries. Experiences from nuclear and aviation industries indicate that mistakes rarely result from neglect, but instead from failures in the systems, processes and procedures. This situation is also the case for health care, as concluded in the foundational report ''To Err is Human: Building a Safer Health System'' published by the Institute of Medicine (IOM) in 2000 (2). This report urged healthcare organizations to implement initiatives to improve safety and represented a milestone in the establishment of safety culture in medical settings. The report "IRPA Guiding principles for establishing a radiation protection culture" published in 2014 acknowledged that additional factors should be considered with regards to radiation protection culture in the medical field (3). An international initiative was launched in 2015 to join the efforts of IRPA, IOMP, IAEA and WHO to particularly focused on radiation safety culture in health care (RSCHC). The intention was enabling synergies by combining and complementing the respective roles and mandates of these organizations thus expanding the scope of stakeholders to be engaged to achieve greater impact. The ultimate purpose was to develop a framework for enhancing safety culture in the medical use of radiation, including the provision of guidance and tools. As a first step, feedback from relevant stakeholders was collected during a series of international workshops held in different regions of the world. Six regional workshops of RSCHC were held in (1) Latin America (Buenos Aires, Argentina, April 2015); (2) Europe (Geneva, Switzerland, December 2015); (3) Africa (Stellenbosch, South Africa, November 2016); (4) Middle East (Doha, Qatar, February 2017); (5) Asia (Kuala Lumpur, Malaysia, November 2017) and (6) North America (San Diego, USA , February 2019). A major challenge at the beginning of the regional workshops was to clarify the concepts of safety culture vs. radiation safety and build a common understanding about radiation safety culture. These meetings provided a platform for mapping stakeholders, suggesting key elements of a framework for a sustainable RSCHC, discussing the strengths, weaknesses, opportunities and threats for enhancing RSCHC in the respective regions (i.e. SWOT analysis), developing strategies for engaging patients, families and communities, proposing tools for strengthening RSCHC and identifying indicators for assessing RSCHC. Building upon stakeholders' feedback, a guidance document for health care providers was developed and will be presented in this manuscript. It proposes a framework for enhancing radiation safety culture in medical facilities, learning from safety culture experiences in other areas, suggests tools for establishing and maintaining RSCHC, discusses approaches for the qualitative and quantitative assessment of RSCHC, provides examples of good practice, includes a glossary and a list resources for further learning.

Keywords: safety culture, radiation, medical exposure, health care

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PS4 (T4.D-0587)

Assessing local patients' and medical staff knowledge and awareness of radiation dose and risks associated with medical imaging : A survey study

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Exposure to high levels of radiation increases the risk of cancer. Some authors believe that even low levels of radiation such as those encountered during radiological examination lead to increased cancer risk. Others argue that radiation estimates are overstated. In our study, we aimed to assess the awareness of radiation dose and associated risks and to evaluate degree of knowledge among local patients, paramedical and medical professionals.
To reach this goal, we performed a retrospective study through a 20 question-survey distributed on social tunisian networks and conducted orally in the trauma center of Ben Arous for patients and hospital staff. 387 survey were completed. Our study population included 3 groups.
In patients' group (n= 122), population represents a highly educated group of people with 58% and 20% having attended university and high school respectively. 19% of patients admitted that they don't make differences between ultrasonography, MRI, CT and plain X-ray. 91% of patients confirmed that a woman should tell the radiologist about pregnancy and only 60.7% think that an X-ray is forbidden in pregnant women. 73.5% of patients revealed that doctors who prescribed the CT didn't inform them that it exposes to radiation by X-rays.
In paramedical group (n= 31), 68% revealed that they are worried about radiation. 3% and 29% were not aware about the radiation-free nature of ultrasonography and MRI respectively. 90% of paramedics well believe that we should minimize irradiating examinations in children compared to adults. 64.5% would prefer to undergo a more definite and more radiating test if they had an illness to diagnose than a less definite and less radiating test.
In medical group (n=234), only 44.5% were worried about radiation. 30.3% et 32.9% wrongly believe that received dose on a chest CT is respectively 5 and 10 times the received dose on a chest X ray while only 12.5% correctly believe that it's 200 times actually (Table 1). 72% thinks there are not precautions to take when irradiating children and only 28% truly believe there are such precautions. Among them, only 23% and 12% knows that respectively external genitalia should be protected .64% of practicians don't warn their patients about X-Ray exposure when they prescribe CT. 36% would undergo a CT if they are offered to make a CT scanner for free and not indicated for their condition.
The degree of knowledge and awareness of X-ray irradiation in medical imaging and its effects is not satisfactory in the group of medical and paramedical personnel and low in the group of patients. It is necessary to offer safety culture training for new hires as part of their overall welcome. It is mandatory to strengthen the training of hospital staff by reminding them of the principles of radiation protection and to inform patients about the nature and risks of X-ray irradiation.

Table 1. Assessing awareness of radiation dose of a chest CT scan among the medical group

Radiation dose received during a chest CT scan compared to the dose received on a chest x-ray is	Inferior	Equivalent	5 times	10 times	200 times
	n= 24 (10,3%)	n= 33 (14%)	n=71 (30,3%)	n= 77 (32,9%)	n= 29 (12,5%)

Keywords: Radiation protection, Radiology, Safety culture

PS5 (T5.1-0522)

Recycling of Phosphogypsum as TENORM

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Phosphogypsum (PG) is the waste product from processing of natural phosphate deposits with sulfuric acid to extract phosphoric acid for fertilizer production. This processing has been operating worldwide for many decades for fertilizer production and has produced enormous volumes of PG tailings material worldwide. In many European countries such as Greece, Bulgaria, Serbia, Kosovo, but also in US, China and countries in North Africa like Morocco where the phosphate industry forms a backbone of the national economy, the PG industry is landscape-forming and often occupy coastal regions which could otherwise be utilized for generating alternative income. Recycling and recovery of these areas currently covered by PG tailings material would allow reclaiming these regions for agriculture and tourism.

One critical issue of PG tailings is their radionuclide content (mainly Radium 226). PG tailings material is generally classified as Technologically Enhanced Naturally Occurring Radioactive Material (TENORM). Due to the TENORM classification of PG material, remediation of PG tailings has become a severe environmental concern of all stakeholders involved. National governments are concerned about the enormous costs estimated for PG radiological remediation, and the public is increasingly concerned about possible growing environmental and health hazards originating from PG tailings worldwide. Based on the anticipated growth in global population and associated demand for more nutrition, all stakeholders anticipate an increasing demand for (phosphate-based) fertilizer and therefore a successive growth of PG tailings volume.

On the other hand, gypsum becomes more and more a short resource, as the production of gypsum by the operation of the off gas cleaning system of coal-fired power plants will be dropped out in future. The industry for construction material is searching for new sources of gypsum for the next decades and is highly interested in using phosphogypsum also for the production of construction material.

However, the radionuclide content of PG tailings (mainly Ra 226) is not homogenously distributed within the typical PG tailings. Preliminary PG tailings assessments indicate that many PG tailings sites contain little critical radionuclide content (i.e. material above the allowed legal environmental radiation limits). Other PG sites contain “layers” of “hot spots” of Ra 226-bearing PG. Sorting and separation of these critical, radionuclide-bearing material from the non-critical PG material would allow to reduce and minimize the radiologically active PG volume due for radionuclide remediation, and allow recycling of the non-critical PG portion for the construction industry.

An automated and mobile system developed by DMT et al. allows efficient classification and sorting at 5 to 8 t/h of even large tailings volumes. Classification and sorting of NORM-bearing (mainly Ra 226) tailings material will separate Ra 226-bearing PG in need of radiological remediation from uncritical NORM-free PG material which can be recycled without radiological remediation into gypsum-based construction material. This project financed by EIT Raw Materials is titled raPHOSafe.

PS5 (T5.1-0657)

Development of Technical & Scientific Capability for Enhancing NORM Regulation

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The National Nuclear Regulatory in South Africa has established the Centre for Nuclear Safety and Security as a technical and scientific support organization, to develop and enhance capacity and skills in nuclear & radiation safety and security nationally and regionally. The Centre collaborates with local and international academic institutions to conduct scientific research related to nuclear safety in order to improve the regulatory practices and processes in South Africa. To date, the regulatory body does not have a defined regulatory framework or reference standards to protect the public against risks related to radiation exposure from radon and radioactivity in drinking water. Therefore, the development of technical and scientific research programmes relating to this would be of importance.

South Africa is known for the historic extraction of gold and uranium, resulting in mine tailings, which contain significant quantities of uranium and its radioactive decay products. Most of these waste products are now abandoned and disposed of in an open space in the form of tailings, therefore, leading to the readily dispersal of radioactive metals through wind and water erosion. The main concern South Africa is facing is that the tailings are located proximal to densely populated residential areas. Also, some areas depend mostly on groundwater that could be impacted by radioactive elements released from nuclear mining activities and waste products. Under such situations, the exposure to uranium and its progeny may be substantial thus hazardous to human health. Considering the consequence of such environmental and health issues caused by the radioactive elements, it is therefore necessary to develop education and training program for young researchers in South Africa. Currently, the Centre is conducting research focusing on radon, levels of radioactivity in soil and water in the vicinity of uranium mines. The outcomes of the research will result in a number of benefits, which, amongst other will include: (1) Better understanding of the nature and extent of radiation contamination. (2) Development of national regulatory standards aligned with requirements from the ICRP and IAEA. (3) Improve the public awareness on radiation exposure and its associated risks. (4) Expand the research capabilities of graduates and researchers. (5) Provide scientific evidence as a basis for policy and decision-making relating to the protection of the public and the environment against radiation exposure.

Keywords: *NORM, Radon, Radiation Exposure*

ACKNOWLEDGMENTS

University of Pretoria, Department of Mechanical & Aeronautical Engineering for hosting the CNSS

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PS5 (T5.1-0717)

Estimation of Radiological Impact from Uranium Mining and Milling Operations at Mkuju River Project in United Republic of Tanzania

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The purpose of this study was to estimate the radiological impact to the public from the mining and milling activities at Mkuju River Project in Namtumbo district in Tanzania. It was assumed that these activities caused exposure to the public due to ^{238}U and ^{232}Th in uranium ore. The assumption was that the main cause of exposure were the dust from the mining and milling activities and these dust were transported through the atmosphere. The Gaussian Atmospheric Model was assumed to be responsible for transferring the dust to the receptor locations. The exposure pathways assumed were external, ingestion and inhalation of ^{238}U and ^{232}Th and their daughter radionuclides. MILDOS-AREA computer code was used for analysis and estimation of the public exposure dose. The input source term were taken from the previous study done at the site as part of the baseline study for determination of natural radioactivity around the site. The site specific parameter and meteorological data were entered into the code. The dose were analyzed as individual and population dose, and the area considered to be affected by the plume was the area within 80km from the source. It was observed that the pathway that contributed much for both individual and population exposure was inhalation, contributing 4.88E-04 mSv for individual and 3.31 mSv for population exposure. The individual total receptor pathway dose was 5.16E-04 mSv while total population pathway dose was 5.27 mSv. However, the dose for both categories did not exceed the dose limit of 1 mSv per year for the member of public. It was concluded that there is no radiological risk for the community living in the vicinity of Mkuju River Project as a results of mining activities. The results of this study are summarized on the table below.

Table 1. Individual and population doses from different pathways (mSv)

	Individual Dose	Population Dose
Groundshine	1.92E-07	1.07E-03
Cloudshine	1.17E-05	7.88E-02
Inhalation	4.88E-04	3.31E+00
Plant Ingestion	1.33E-05	-
Meat Ingestion	2.59E-06	-
Milk Ingestion	7.29E-07	-
Total	5.16E-04	5.27E+00

***Keywords**: Radiological Impact, Gaussian Plume Model, Uranium Mining*

ACKNOWLEDGMENTS
This research was supported by 2019 Research Fund of KEPCO International Nuclear Graduate School (KINGS), Republic of Korea.

PS5 (T5.1-0774)

Quick-Erect Stopping System (QESS) for Miners to Prevent High Radon Exposures

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A new Quick-Erect Stopping System (QESS) for miners was developed to reduce the radon exposures of miners are employed in remediation work at old mining sites in Germany (Dehnert 2020).

The QESS is a light-weight, modular, and reusable construction kit consisting of interlocking telescopic aluminum tubes, radon-proof foil, and expanding foam to shut off radon-rich parts of galleries of any width within a few minutes only.

The construction kit contains 15 two-piece, extendable telescopic tubes in four different lengths. All tubes have regularly placed slots on both sides from top to bottom. The tubes have tips made of hard plastic at both ends. The tube tips can be inserted into the slots of other tubes. One of the two tips of a tube is equipped with a spring. Therefore, the telescopic tubes can be pulled out and stretched. The tubes have two screws to lock the stretched parts. Finally, the tubes are color-coded with reflecting foil (red, green, yellow, and blue) indicating their length (Dehnert et al. 2019).

Setting up the QESS takes two miners 15 to 30 minutes. The QESS can be dismantled in less than five minutes. The QESS is reusable except for the foil and the foam and can be re-erected, for example, after a blasting operation at the same location or reused at another location.

The QESS was tested in the Reiche-Zeche Research and Teaching Mine of the Technical University Bergakademie Freiberg, in the Edgar Experimental Mine of the Colorado School of Mines and at some small underground construction sites of old mining in the Ore Mountain (Erzgebirge).

Fig. 1. Quick-Erect Stopping System (QESS) in the Edgar Mine, Idaho Springs, Colorado

Keywords: radon, mining, ventilation

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PS5 (T5.3-0385)

Assessment of geopolymer material for the manufacture of a nuclear fuel core catcher

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The aim of this study was to develop and structurally evaluate ordinary Portland cement (OPC) and geopolymer matrices and to use the OPC results as baseline standard for the envisaged geopolymer species to attempt to supersede current OPC core-catching application. This was done by assessing the durability of different OPC matrices in comparison with various geopolymer matrices when exposed to the ASTM standard; water absorptivity, sulphuric acid resistance and compressive strength testing protocols. In this study the water to cement ratios (w/c ratios) for ambient cured OPC baseline pastes were varied between 0.31-0.40 and were subjected to air, water and plastic sealed curing. OPC samples cured in sealed plastic bags with a 0.31 w/c ratio resulted in optimum compressive strength of (30.94 (33) MPa), a low percentage weight (% wt.) loss due to sulphuric acid digestion (83.95 (33)%) and lower sorptivity rates relative to other OPC matrices. For the second part of the investigation, it was observed that contrary to OPC samples, plastic sealed cured fly ash based geopolymers (F-GIP) activated with 14 M $NaOH/Na_2SiO_3$ solutions, and with a higher AS/B ratio (0.40) resulted in an optimum compressive strength of 35.1(30) MPa, an insignificant degree of % wt. loss when exposed to sulphuric acid (1.28 (11)%) and low sorptivity rates. According to this study the F-GIP mixtures significantly outperform the OPC idealized baseline matrices for the proposed application as a core-catcher material.

Keywords: *Water absorptivity, Sulphuric acid resistance, Compressive strength,*

ACKNOWLEDGMENTS
The financial assistance of the National Research foundation (NRF) towards this research is hereby acknowledged. Opinions expressed and conclusions arrived at, are not necessarily to be attributed to the NRF.

PS5 (T5.3-0696)

TERRITORIES WP1 - Quantifying variability and reducing uncertainties when characterizing radiological exposure of humans and wildlife by making the best use of data from monitoring and of existing models

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TERRITORIES project, part of the CONCERT funded European project, was launched in 2017 and ended in December 2019. Whithin TERRITORIES the working package 1 (WP1) was in charge of investigating the possibilities of reducing the uncertainties inherent in every radioecological model and in the monitoring of physical quantities. In order to achieve this aim, efforts were put in the improvement of the already existing models and their test against measurements compiled from several real existing contaminated sites.

- A first task was the development of a database, the so-called Territories Libary Database (the TLD), with measurements around five contaminated sites (a NORM contaminated site in Norway, a NORM contaminated site in Belgium, a NORM contaminated lake in Poland, 137Cs contaminated forests in Fukushima and artificial radionuclides contaminating Sellafield coasts). The TLD is publicly available.
- Two theoretical reports providing guidance on the characterizing together with methods to reduce some components of sampling, monitoring and modeling the behaviour of the radionuclides in the environment were produced. To be mentioned the focus on sampling uncertainties and the uncertainties associated with the conceptual model. Recommendations included in these two reports will be provided.
- Finally a method was develop to quantitatively score the improvement of any new model against any existing model, in the base of their comparison against measurements, together with the application of the model. Eleven different fit-for-purpose models were tested against every site in the TLD. Recommendations on how to develop a model and show the improvement of the more advanced models is shown.

All the results of the TERRITORIES WP1 will be presented.

PS5 (T5.3-0696)

TERRITORIES WP1 - Quantifying variability and reducing uncertainties when characterizing radiological exposure of humans and wildlife by making the best use of data from monitoring and of existing models

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TERRITORIES is part of CONCERT. This project has received funding from the Euratom research and training programme 2014-2018 under grant agreement No 662287. This publication reflects only the author's view. Responsibility for the information and views expressed therein lies entirely with the authors.
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EUROPEAN JOINT PROGRAMME FOR THE INTEGRATION OF RADIATION PROTECTION RESEARCH

CONCERT TERRITORIES

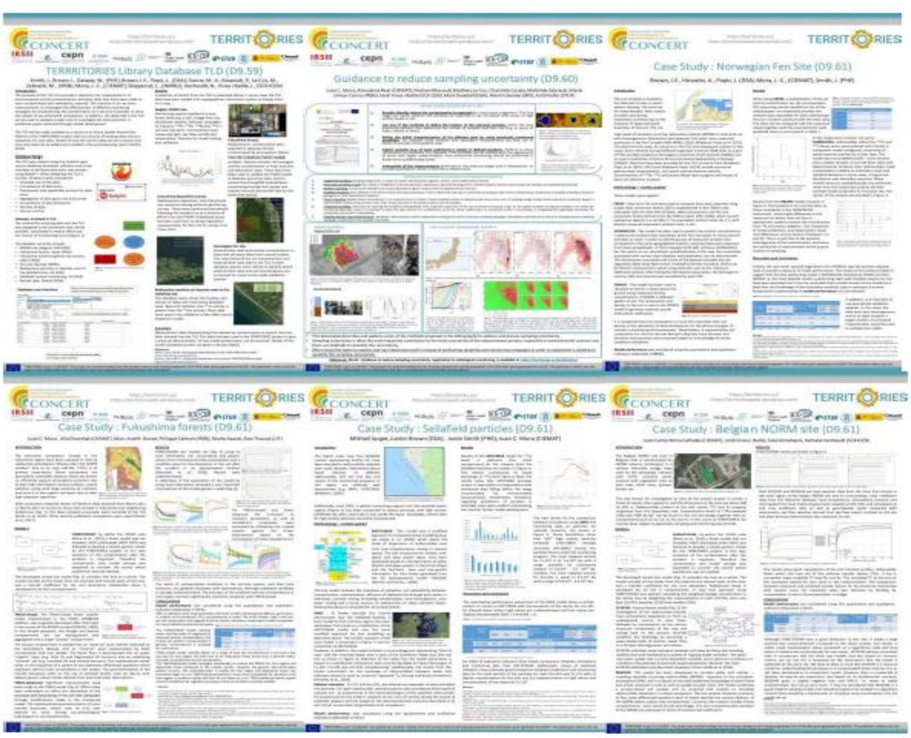

Figure 1.- Posters presented as part of the final event, held in Aix en Provence (France) in November 2019. A summary of the more important results presented there will be shown in the IRPA Congress.

PS5 (T5.3-0819)

Applicability Evaluation of Gaussian Dispersion Model for Atmospheric Radiation Dispersion from the Hanul Nuclear Power Plants with Complex Terrain

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Environment radiation evaluation through atmospheric dispersion modeling is intended to assess the impact of radiation from a nuclear power plant (NPP) on residents living nearby the plant. The Korean Nuclear Safety and Security Commission (NSSC) Notification No. 2017-26 concerning the assessment standards of the meteorological conditions of nuclear facilities requires that radioactive atmospheric dispersion and dilution should be evaluated using appropriate evaluation model for the topographic and climatic characteristics in the regions.

The Korea NPPs have applied XOQDOQ and PAVAN, which use conservative factors based on Gaussian plume model to assess public radiation dose during normal operation and accidents. Gaussian plume model is originally designed in the plat terrain and constant wind conditions. Based on the NSSC regulations, the Korean nuclear regulatory body required the Korea Hydro & Nuclear Power (KHNP), the NPP licensee, to verify the applicability of its dispersion model under the complex topographic and climatic conditions of the domestic NPPs.

In response, the KHNP Central Research Institute (CRI) studied and analyzed the relevant tests conducted at home and abroad, technical review of actual data collection methodology (field test, tunnel test, turbulent measurement, activity concentration measurement, etc.) to establish how to perform the verification for applicability of atmospheric dispersion coefficient for calculating public dose. Based on the prior research, the experiment was designed to conduct eight field tracer experiments in a year to verify the conservatism of the current atmospheric dispersion model for the Hanul NPP site.

The main experimental steps are as follows:

1) Configure tracer release and sampling points (about 150 points): release at one point in the NPP site and sampling around the site with duplex line (at a radius of about 1 and 2 km from the release point)
2) Tracer gas(SF_6 gas) continuous-release (120 min.) and air sampling (at intervals of 10 min. over 1 hr)
3) Air sample collection and tracer gas concentration analysis
4) A comparative analysis of the tracer gas concentration analysis results and modeling results

The concentration analysis results from the tracer experiments were generally several to hundreds of times smaller than the modeling values being used in NPPs. Thus, it is concluded that the current atmospheric dispersion evaluation method based on the Gaussian plume model is more conservative than the actual atmospheric dispersion of radioactive materials, and it is suitable for securing the radiation safety of residents. This result is used to resolve the outstanding issues associated with Construction and Operating Licenses and to improve the safety of NPPs under operations.

***Keywords**: Atmospheric dispersion, Gaussian Plume Model, Field tracer experiment*

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PS5 (T5.3-0924)

Automated Gamma Isotopic Analysis module for EPRI SMART Chemistry Project

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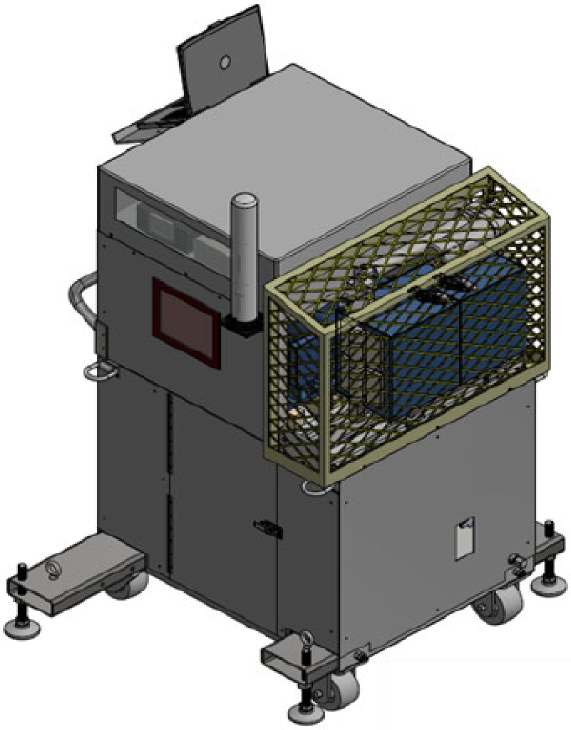
Measurement of gamma isotopic activity in the primary coolant of an operating Nuclear Power Plant [NPP] is a routine operations task. These results are used to monitor and trend for fuel reliability and asset management. Today this task is primarily performed by manually extracting samples of the fluid in the plant, taking them to the assay laboratory, dispensing the proper amount in to the assay container, and then counting on a HPGe gamma spectroscopy system. This process is time consuming for the NPP staff, creates risk of spills of radioactive fluid, and exposes the staff to radiation fields. Furthermore, potentially useful information from short-lived radionuclides and dynamic activity changes between the discrete sampling periods is lost.
EPRI decided to incorporate an inline HPGe gamma isotopic analysis module in its SMART Chemistry system to enable automatic and real-time gamma isotope monitoring and trending of the primary coolant. The developed design is a stand-alone module capable of processing the reduced pressure and temperature primary coolant either independently, or as received from the SMART Chemistry sample conditioning skid. The module is comprised of coolant handling system, the detection system, and the data handling and analysis platform. The included Figure shows a 3D rendering of the system currently under construction.

The coolant handling system in the bottom cavity automatically monitors flow and temperature, controls the dual sample assay process, controls the filling process for the assay chambers, and controls the process to flush the chambers and to measure background.



The detection system is a shielded HPGe detector, optimized for a wide dynamic counting range. A unique aspect of the system is the ability to perform a continuous sequence of assays on a continuously flowing sample, or to extract a sample and institute a decay period before counting, or to continuously alternate between both types of measurements. In the alternating mode, for example the system would normally doing continuous assays, then once every 24 hours would count a sample that has decayed for 24 hours, then return to the continuous mode. Another unique aspect of the system is the ability to simultaneously execute analysis with multiple counting times on each sample; short count times to record dynamic changes in activity, in parallel with long count times for lower detection limits.

The data handling system consists of a remote spectral and results display and relational data base system storing all of the raw data and analyzed data.

This paper will describe the system in more detail and show factory testing results, prior to deployment.

Keywords: *Primary Coolant, Automated Gamma Spectroscopy, HPGe spectral analysis*

Poster Session T5

PS5 (T5.3-0979)

Three-dimensional numerical groundwater flow and advective transport modeling of the Thyspunt area Eastern Cape, South Africa

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A conceptual hydrogeological and numerical groundwater flow modelling study is being undertaken around and within the proposed ESKOM Thyspunt Nuclear Site, located 120 km west of Port Elizabeth. The area is characterized by folded and jointed complex geological conditions. The local geology comprises the Table Mountain Group (TMG) and the Bokkeveld Group of the Cape Supergroup. These geological units are unconformably overlain by Quaternary to recent sand deposits of the Algoa Group. The main objectives of the study are to improve the understanding of the prevailing hydrogeological condition through a conceptual hydrogeological model and numerical groundwater flow model, and thereby to independently validate previous studies undertaken around the Thyspunt area. Thus, a robust conceptual hydrogeological model is developed through detailed aquifer characterisation including pumping test analyses, determination of groundwater occurrence, storage, flow, hydrogeochemical and environmental isotope analyses. The area receives a mean annual precipitation (MAP) of around 622 mm/a, of whichabout 6% recharges the groundwater. Groundwater occurs within intergranular aquifers of the Algoa Group and fractured quarzitic aquifers of the TMG. The depth to groundwater ranges from 4.5 to 28.9 m below ground level (bgl) and though the local groundwater flow is complex, the general groundwater flow direction is from west to east, towards the Indian Ocean. The upper unconfined intergranular Algoa Group aquifer and the deeper semi-confined fractured TMG aquifers are characterised by wide ranges of hydraulic properties, including aquifer thickness (2.2 - 22.0 m and 18.0 - 138 m), hydraulic conductivity (4.5 - 19.1 m/d and 8.9×10^{-3} -1.58 m/d), transmissivity (108.3 - 275 m^2/d and 0.4 - 44.0 m^2/d), specific yield (1.5×10^{-2} - 0.1) and storativity (5.0×10^{-5} - 5.9×10^{-3}), respectively. Groundwater and surface water are characterised by circum-neutal pH condition, with mean pH of 7.37 and 7.67 and mean Electrical Conductivity of 1103 μS/cm and 1731 μS/cm, respectively. The main hydrochemical facies of groundwater in the shallow Algoa Group aquifer is Ca-Mg-HCO_3 type and groundwater circulating in the deep TMG aquifers are mixed Ca-Mg-Cl and Ca-Mg-HCO_3 types. Environmental isotope signatures (δ^2H and $\delta^{18}O$) indicate groundwater - surface water interactions. The three dimensional numerical groundwater flow model is constructed based on the conceptual hydrogeological model using MODFLOW within the GMS graphic user interface. The MODFLOW-NWT, a newton formulation for Modeflow-2005 that uses the UPW-Upstream weight flow package and NWT-Newton solver was used. The preliminary modelling results indicate near pre-development conditions for the proposed nuclear site.

***Keywords**: Environmental Isotopes, Numerical model, Eastern South Africa*

PS5 (T5.3-1070)

Preliminary evaluation of public radiological risk reduction using pool scrubbing

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Reducing the amount of release of radioactive materials during severe accidents has been addressed as an important issue since the Fukushima accident occurred in 2011. In the event of a severe accident that the nuclear fuel melts, the fission products behave in aerosol form by mixing with steams and non-condensable gases inside the containment and can be released out of the containment. Radioactive material released into the environment may pollute not only the atmosphere but also increase public radiological risk. Therefore, appropriate countermeasures are needed to reduce this radioactive material emission. As a way to reduce this risk, there is a method of reducing the leakage of radioactive material by using pool scrubbing. Pool scrubbing is a phenomenon of removing radioactive material using the water pool. When aerosol containing radioactive material is injected into the water pool, the radioactive particles suspended in the rising bubble move to the bubble interface by gravitational sedimentation, Brownian diffusion, inertial impaction, thermophoresis, and diffusiophoresis and particles are collected by water. In other words, in pool scrubbing, water acts as a filter and deposits the radioactive particles at the bubble interface. In this research, the self-developed code(POSCAR: Pool Scrubbing Aerosol Removal code) was used to predict the decontamination factor due to the pool scrubbing process. This code was developed taking into account the thermal-hydraulics behavior of bubbles and various aerosol removal mechanisms. In addition, IMBA(Integrated Modules for Bioassay Analysis), were used to evaluate the internal dose of the human body. Using aforementioned codes, the public radiological risk was evaluated by comparing the internal doses assessed by IMBA before and after pool scrubbing.

Keywords: *Severe accident, Pool scrubbing, Radioactive aerosol removal*

ACKNOWLEDGMENTS

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PS5 (T5.3-1149)

The dose assessment for OPR1000 during Loss of Spent Fuel Pool Cooling

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Loss of Spent Fuel Pool Cooling (LOSFPC) is the accident that the cooling system do not operate and the water level is decreased due to the decay heat released from the stored spent fuel assemblies in Spent Fuel Pool (SFP). Therefore, most important factor related to LOSFPC is the time interval until the top of the fuel assemblies is uncovered. If the water level above the fuel assemblies reaches a height of approximately 10ft above the top of the fuel assemblies, either due to leakage or boil-off, the habitability of fuel handling area may become an issue. Further, if the water level decreases below the top of the fuel assemblies, significant fission product releases could occur. In this reason, there is a strategy to inject water into the SFP for the mitigation of LOSFPC. The purpose of injecting water into the SFP is to recover SFP water level. This would recover fuel cooling and radiation shielding in a short time. In this paper, alienated dose assessment for the evaporated radioactive material was performed while the water level reaches above the fuel assemblies. And, the OPR1000 was selected for reference plant. First, the assessment of time to reach the boiling temperature and the water level corresponding to the top of fuel assemblies was performed. As the assessment results, the time to reach the boiling is minimum 4.2 hours and maximum 10.6 hours. And, the time to reach the water level corresponding to the top of fuel assemblies is minimum 36.7 hours and maximum 84.4 hours. Second, the maximum evaporation rate was calculated. The maximum evaporation rate due to the decay power was 94.6 gpm. Based on these results, the dose assessment was performed. For the conservative assessment, the minimum time to reach the water level corresponding to the top of fuel assemblies was applied. Additionally, it was also assumed that the mitigation strategy to inject water into the SFP is not implemented. The specific activity data in coolant of SFP was referred to the specific design document for OPR1000. As a preliminary assessment result, the effective dose is about 0.22 mSv during the LOSFPC accident and this result meets the requirement for acceptance criteria.

Keywords: *Spent fuel pool, Loss of cooling, dose assessment*

PS5 (T5.3-1249)

A framework to understand and model the dynamics of safety management in the operation of a nuclear reactor

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This paper presents a conceptual framework proposal to understand and model the elements and dynamics of the radiological and nuclear safety management of a nuclear reactor operating organization. The methodology to develop this proposal is based on a systematic bibliographic review of the topics of safety management and accident causation theories for the nuclear industry (Acuña et al. 2020) and the inductive-deductive application method to the findings and expert judgment.

The deductive method is applied based on the definitions (premises) on safety and risk raised in any human activity context. On this basis, this paper proposes to revisit the purposes of safety and risk in the context of the nuclear industry. The inductive method is applied based on the Man, Technology, Organization, Information framework (Wahlström and Rollenhagen 2014; Wahlström 2018) and the contributors to the risk of modern nuclear accidents.

Finally, and regarding the day to day of a real plant and the dynamics of his safety, the states of these socio-technical systems evolve (drift, change) over time. Their instantaneous state is continuously affected by the drift of the system. This paper identifies the fundamental units of a dynamic model and its architecture (see Figure 1). On this basis, a new framework is concluded and proposed that aims to integrate these contributors (technological, human, and organizational).

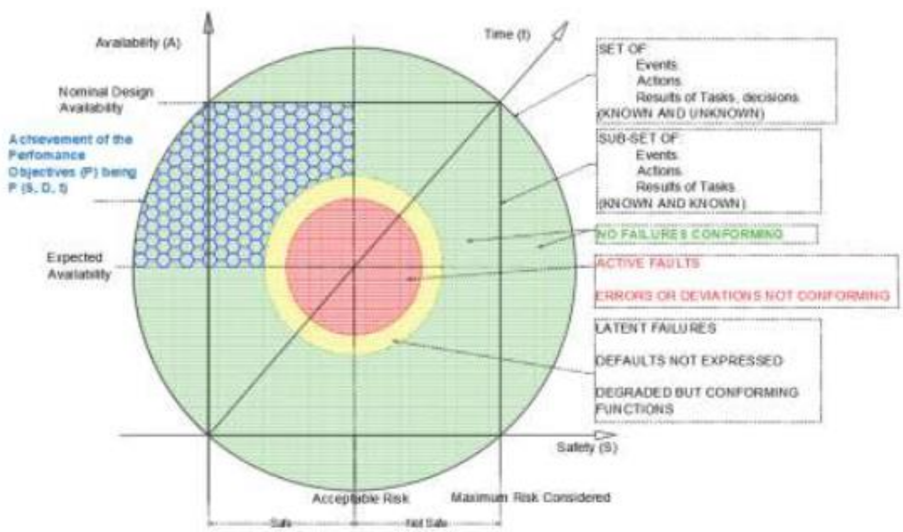

Fig. 1. Safety integrated domains

Keywords: *radiological, nuclear, safety.*

PS5 (T5.4-0045)

Dose Assessment to Radiation Workers from Accidents in Industrial Gamma Radiography using Monte Carlo Method

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Industrial radiography resulted in higher radiation exposure in case of a bad practice or in an accident. Many factors were responsible for accidents in industrial radiography including inadequate regulatory control, failure to follow operational procedure, inadequate training, inadequate maintenance, human error, equipment malfunction or defect, design flaws and even willful violation. The purpose of this study was to assess the radiation dose to workers from potential hypothetical accidents in industrial gamma radiography of United Republic of Tanzania using Monte Carlo method. The accident was simulated using industrial gamma radiography source of ^{192}Ir with the activity of 100 Ci which was average usage in Tanzania. Visual Monte Carlo (VMC) dose calculation software was used for the estimation of effective doses and organ doses to the radiation worker using ICRP male phantom to represent human body. Two exposure durations were compared with 10 minutes and 30 minutes. The results of this study were summarized as shown in Table 1 for only those organs with relatively high doses. From the exposure for 10 minutes, the results were found to be below the ICRP recommended dose limit for occupational exposure which was corresponding to maximum dose of 50 mSv in any year over 5 years. But, the effective dose was found to be higher than the ICRP recommended dose limits in the case of exposure for 30 minutes. The equivalent dose to skin was estimated to be approaching the ICRP recommended dose limit of 500 mSv/yr. The equivalent dose to lens of eye was found to be 9.48 mSv/yr. Although these doses were lower than the recommended dose limits, they represented higher doses for a single exposure scenario. Based on the exposure duration and the activity of gamma source, this exposure scenario could lead into any deterministic health effect, and therefore handling of industrial gamma sources should adhere to operational procedures, training, maintenance, regular checkup of devices and regulatory control.

Table 1. Radiation doses (mSv) for two exposure durations

	10 min.	30 min.
Effective dose	26.76	80.29
Equivalent dose to the skin	123.93	371.79
Equivalent dose to the bone surface	118.21	354.63
Equivalent dose to the testes	136.03	408.09

Keywords: Industrial gamma radiography, Dose assessment, Monte Carlo

ACKNOWLEDGMENTS

This research was supported by the 2019 Research Fund of the KEPCO International Nuclear Graduate School (KINGS), the Republic of Korea.

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PS5 (T5.4-0460)

AnnGro® as a transporter of 65Zn and 54Mn into maize (Zea mays L.) via foliar uptake

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In this study, the effectiveness of AnnGro® as a transporter of zinc and manganese to improve the uptake and distribution of nutrients in maize plants was investigated. ^{65}Zn and ^{54}Mn were used as radio-tracers to follow the uptake of the elements 24 hours after application of fertilizer. The experimental setup was such that there were two control sets, one to determine cross contamination where there was no radiotracer applied and the second set was the experiment control where only the radiotracers were applied, without the AnnGro®. Termination was 24 hours after the foliar application. On termination, the weight of the whole plant was measured. Then the plant was sectioned into leaves, stem and root. The samples were dried and ground into a consistent powder. The activity concentrations of ^{65}Zn and ^{54}Mn were determined using gamma spectrometry. The effect of AnnGro® in the uptake of nutrients was determined in the first experimental set, where the same amount of AnnGro® was applied using different concentration amount of radiotracers and a uniform relationship was observed. In the second experimental set, different amount of AnnGro® was applied using the same amount of radiotracers, it was observed that AnnGro® worked better at 1mL/L, which is the recommended amount of liquid fertilizer according to the manufacturer, than in 0,5mL/L and 2mL/L. AnnGro® furthermore worked better over time; an increase of 22% of nutrients uptake was recorded in the first termination (24 hours) while an increase of 33% in the second termination (14 days after application) was observed.

Keywords: *radiotracers; uptake; bio-nano-transporter*

ACKNOWLEDGMENTS

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PS5 (T5.4-0489)

Optimising the dissolution of U/Al in alkaline solutions and subsequent dissolution of the generated uranium residue in nitric acid

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The aim of this study was to determine the optimum dissolution medium for U/Al material in either NaOH or KOH, and evaluate ion exchange as an alternative purification technology. This investigation was conducted by assessing the dissolution kinetics of aluminium in hydroxide media and further optimising dissolution parameters that would be implemented on an unirradiated aluminium target plate containing depleted uranium (DU). Subsequently, the optimum dissolution parameters for uranium residue in nitric acid and the effect of aluminium on the efficiency of extracting uranyl nitrate using tributyl phosphate (TBP) were determined. This study further assessed uranium purification using resin technology. The optimum medium and dissolution parameters for dissolving Al were 6.0 M NaOH in 3.0 M excess at 80°C, 1000 rpm with added 3.5 M $NaNO_3$; all based on dissolution rate and re-precipitation time. The aforementioned parameters were implemented on a DU target plate and revealed that the dissolution rate of the target in NaOH only was higher (0.5328 g/min) compared to the reaction where $NaNO_3$ was added (0.1488 g/min). The recovered uranium residue was optimally dissolved in 3.0 M HNO_3 25°C based on dissolution efficiency. Solvent extraction was performed to determine the effect that aluminium had on uranium extraction and stripping. The optimum concentration for stripping uranium was detected at 1.0 M $(NH_4)_2CO_3$. Using ion exchange resin, three resins (cation and two chelating) were tested and compared based on their distribution coefficient of uranium with the highest K_D (4461.9 ml/dry g) observed using Amberlite IRC747 resin. A uranium feed concentration of 480 ppm experienced breakthrough at 60.69 BV, equivalent to a breakthrough capacity of 29.11 g U/L resin. Comparing both recovery methods, solvent extraction recovered 86.27% of uranium in the solution in contrast to ion exchange (82.08%). Therefore, solvent extraction is the suggested method for uranium extraction.

Keywords: Aluminium, dissolution, ion exchange (IX), solvent extraction (SX), target plate, tributyl phosphate (TBP)

PS5 (T5.4-0614)

Fabrication and Properties of Gaseous Radiotracer Carrier Based on Hydrquinone Clathrate

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Radiotracers have been used widely in various industries such as the petrochemical, steel, and automobile industries. The selection of appropriate radioisotopes to be used as tracers is very important for confident results, and the selection procedure needs to consider physical and chemical characteristics of process fluids to be traced. In addition, their nature needs to be investigated to minimize the environmental impact of radiotracers during experiments. A gaseous tracer is produced by irradiating target elements with neutrons in a research nuclear reactor. For safety reasons, radioisotopes of a short half-life are preferred as tracers. Because it takes a considerable amount of time to transport radiotracers from a radioisotope production facility to a tracer investigation site, the gas target to neutron bombardment is usually pressurized in a quartz containers. The physical integrity of such a container needs to be guaranteed during neutron irradiation, but it is almost impossible to produce containers of identical physical strength. The reason for this is that a quartz ampoule is sealed by the melting of its inlet with a high temperature flame while gas is liquefied by liquid nitrogen at the bottom of the ampoule. This uncertainty with gas containers can make radioisotope production hazardous. As a substitute for the traditional quartz ampoule as a gas container, a clathrate compound was considered. It was prepared to hold Ar gas of up to several bars, and it was characterized to confirm its feasibility as a gas tracer carrier under various conditions. Gamma irradiation tests show the clathrate composite to be barely influenced by a radiation environment. X-ray diffraction, in-situ high-pressure synchrotron XRD, Raman spectroscopy, and NMR were used to characterize the synthesized clathrate samples quantitatively. In addition, the gravimetric method was used to study Ar release kinetics as a function of both the time and the temperature dependent phase stability of Ar-loaded clathrates.

***Keywords**: Gaseous radiotracer, clathrate*

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PS5 (T5.4-0687)

Modeling of atmospheric dispersion and radiation dose for a hypothetical accident in Radioisotope production facility

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Atmospheric dispersion modeling and radiological safety analysis is performed for public outside radioisotope production facility (RPF) in case of hypothetical radioactive Iodine spilling and leakage from hot cell. Potential human error is expected and column that occupies Iodine may be broken causing it to spill on the hot cell floor. Ventilation system is dedicated to extract dispersed material through dedicated filters before gases expelled outside the facility. Two scenarios are presented in this paper, the first one is prediction the dispersion with good filtration from extract ventilation system, while the other with loss efficiency of filtration components. The spilled radioiodine is the source term, and the HotSpot 3.1, Health Physics code was created by LLNL [1] is used to provide Health Physics calculation tool for evaluating accidents involving radioactive materials to perform the atmospheric transport modeling which is then applied to calculate the total effective dose equivalent (TEDA) in different atmospheric stability classes, and how it would be distributed to human body as a function of downwind distance and radionuclide activity. The adopted methodology uses predominant site-specific meteorological data and dispersion modeling theories to analyze the impact of hypothetical release to the environment from the selected radionuclide and evaluate to what extent such a release may have radiological impact on public.

PS5 (T5.4-0732)

Safety Assessment for the Recharge of the Irradiation Facility Product I in Cuba

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The Food Irradiation Plant (PIA), type Product I from the former Soviet Union, was put in operation in Cuba in 1987, in the Research Institute for Food Industry (IIIA). It was designated mainly for irradiation of foodstuffs for preservation, later it was used to irradiate other types of products. According to the IAEA classification, this is a category II irradiator, panoramic irradiator with dry storage of the radioactive sources. The facility was initially charged with 52 Co-60 radioactive sources with a total activity of 2.5 PBq (67.6 kCi, in 1986).

There is the intention to recover the irradiation capacities at PIA, so new radioactive sources had to be installed. The 52 disused radioactive sources were previously unloaded from the irradiator and temporarily transferred to a reserve pit located in the same facility [1].

An authorization from the National Regulatory Body was necessary to recharge the facility with the new Co-60 sources. The safety assessment, the radiation protection program, operational and safety procedures as well as the emergency plan were prepared and presented to the Regulatory Body to apply for authorization.

The safety assessment included the dose estimations for normal operations and potential doses in accidental situations. The safety analysis used the methodology of risk matrixes [2, 3]. Fourteen (14) postulated events were identified for the different stages of the process, including: the reception of the transport packages in the facility (five Type B packages with the new fifty two Co-60 radioactive sources were received in the facility in a maritime container), temporary storage of the packages with the radioactive sources in the facility previous to the recharging operations, transfer of the containers with the sources inside the irradiator and the recharging operations.

The risks associated to each initiating event were identified, as well as the safety barriers. The undesirable events were evaluated, taking into consideration the safety measures in place to tackle them and the potential consequences of the events. The safety assessment, described in the paper, was presented to the Regulatory Body for evaluation and the authorization for recharging the irradiator with the new Co-60 sealed radioactive sources was granted.

Keywords: safety assessment, industrial irradiation

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PS5 (T5.4-0747)

Operational Radiation Protection at the Experiments at the CERN Large Hadron Collider (LHC)

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The Large Hadron Collider (LHC) is the largest accelerator constructed at CERN and has started operation in 2007. The LHC machine accelerates protons as well as heavy ions up to lead. It is installed in a 27 km circumference tunnel, about 100 m underground. The particle beams collide at the centre of four experiments installed at the LHC; ALICE, ATLAS, CMS and LHCb. The stray radiation from these collisions activates detector components and infrastructure material, with proton-proton collisions at a center-of-mass energy of 13 TeV being the dominant contribution, except for the ALICE experiment that focuses on heavy ion collisions.

After successfully completing Run 2 in December 2018, the LHC entered a more than 2 yearlong shut-down, denoted as Long Shut-down 2 (LS2). During this period, the whole CERN accelerator complex undergoes a series of upgrades to prepare the machines to be able to fully exploit the potential of the High-Luminosity LHC that will start operating after Long Shut-down 3 (LS3) in 2027. Extensive maintenance works as well as several upgrades are performed notably also in the LHC experiments during LS2.

This contribution presents the operational radiation protection aspects of the LHC experiments. The focus will be the implementation of the ALARA process, as it has been adopted at CERN, and the radiological characterization of components to ensure optimized radiation protection.

In addition to global optimization measures such as radiological area classification, radiologically important interventions are optimized individually. A Work Dose Plan (WDP) is established for an intervention and the expected dose rates are either measured or predicted by FLUKA Monte Carlo simulations. Based on the WDP, the different steps of the intervention are then optimized. Once the predictions of the relevant radiological parameters of the interventions, e.g. the collective dose and the maximum individual dose, are established, the appropriate approval workflow is then followed according to the CERN ALARA rules. The application of this optimization process will be shown for a selected intervention.

Since the configuration changes in the LHC experiments during a shut-down can be complex, the SESAME code often has to be used in combination with the FLUKA Monte Carlo simulation code for the prediction of the residual dose rates. The example of the ATLAS JTT shielding will be given to shown such a change of configuration between beam operation, when the activation occurs, and the intervention itself. FLUKA Monte Carlo simulations are also used to establish the radionuclide inventory to ensure the proper classification of the transport of radioactive components. An example of a shielding element in the ATLAS detector will be given to shown the complexity of these tasks.

Keywords: ALARA, Optimization, Monte Carlo simulation

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PS5 (T5.4-0767)

Hitachi-GE's approach and efforts for radiation protection management in nuclear facilities

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Introduction

Hitachi-GE has been engaged in domestic nuclear power plant outage, improvement work and the decommissioning of the Fukushima Daiichi Nuclear Power Plant. Hitachi-GE has been working on radiation safety promotion and improvement of reliability of management as mentioned below.

Hitachi-GE's approach

Concerning worker dose control, Hitachi-GE operate a system for control and management of worker dose called "HIRACS (Hitachi radiation control system)" that we have developed. HIRACS enables integrated management of information about worker dose, periodic radiation safety training and medical checkup in the same system. As to the approach in the nuclear power plants, we produce "Dose Rate Map", a floor or area diagram colored by dose level to indicate dose rate distributions in the radiation controlled areas. Workers are able to know dose rates in their work place before working.

Regarding an awareness-raising activity, Hitachi-GE provide various training such as, the experience-focused simulation training that be delivered particularly to less-experienced workers. We confirmed that workers have increased levels of their understanding about radiation with the practical methods we devised in order that workers can recognize radiation in a quantifying and simulating way.

As to a daily-basis precautionary and preventive action to workers, we issue and circulates a document sheet by work that details instructions and directions. The sheet specifies radiation dose rates and precautions in the work places. Previously, workers needed to fill out a number of such documents manually and hand-carried hard-copies to all relevant parties for review and approval. Now such workflow has been digitalized that enables online document generation and review-approval transaction at an individual desk. This system has significantly reduced time spent on generation of documents and facilitated data storing and searching compared to the previous process.

By way of promotions and digitalization to improve work efficiency, we have been developing a tablet device that is linked to a survey meter to streamline and facilitate our radiation survey for the past few years. We currently transcribe a survey record manually from a draft hand-written record created in a work place of the radiation-controlled areas when returning to the office. Linking a tablet device and a survey meter makes it possible to complete producing survey records in a work place. This new device is expected to avoid incorrect entry due to human-error in addition to saving time spent on transcription.

Conclusion

Hitachi-GE has performed sustainable and robust worker dose management and achieved maintenance of safe work with the various efforts as mentioned above. We will continue to develop various systems for work efficiency and aim for the best practices.

Keywords: *Radiation, Safety, Reliability*

PS5 (T5.4-0791)

Development of risk assessment tool using the Excel VBA; For non-destructive testing (NDT) field

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Radiation is used in various fields such as medical, industrial, and research fields. From the analysis of the exposure dose of radiation workers by each field that is used radiation for the last 5 years, we confirmed that the exposure dose in non-destructive testing field is the highest [1]. However, we generally assess the risk of the field only as exposure dose regardless of characteristic of each field. Therefore, we want to develop the excel tool to assess the risk of non-destructive testing field using the various factors including the exposure dose.

In the previous study, we suggested the items for risk assessment in non-destructive testing field through the questionnaire and experts' advise [2]. The items are as follows; 1) Radiation source, 2) Exposure dose, 3) Workplace management, 4) Workers with personnel dosimetry problem, 5) Periodic regulatory inspection. On the basis of this result, we made the excel file that presents the model for risk assessment with the input of each item of non-destructive testing company. The process of the risk assessment of non-destructive testing company is as follows.

1) We confirm the average of all company and the figures of each company by each factors.
2) To derive the final company score, the figures of company divides by total average and multiplies the weight factor (Weight factor is taken into account only for pentagons.)
3) Based on the final company score and total average, we presents the model and safety grade.

During the development of excel tool, there were the following problems;

1) When we evaluate the risk of NDT companies, we should input the value of all items every time,
2) We should make all files that contain the risk assessment of each non-destructive testing company,
3) When we perform the tracking analysis, a lot of storage space is required.

To solve this problem, we developed the following tool using VBA code in excel. The main functions of the tool are as follows;

1) When we select the company, the program brings the data for selected company information,
2) The program presents and deletes the pentangular/heptagon/decagon models,
3) The program presents company safety grade and comprehensive safety grade according to model.

Excel tool is shown as Fig 1. It is expected that we can use the excel tool developed through this study as supplementary data for risk assessment by each non-destructive testing companies.

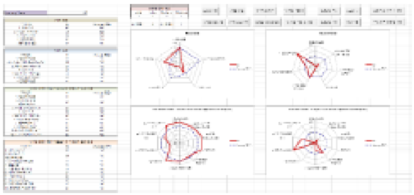

Fig. 1. Risk assessment tool for non-destructive testing (NDT).

Keywords: Non-destructive testing (NDT), risk assessment model, Macro & VBA

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PS5 (T5.4-0808)

Overview of the DOE-NNSA Office of Radiological Security (ORS) and Support Assistance in Implementing Machine-Based Irradiation Solutions

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The mission of the U.S. Department of Energy's Office of Radiological Security (ORS) is to prevent the use of radioactive materials, otherwise intended for applications that are peaceful and beneficial to society, in acts of terrorism. The ORS mission implements a holistic approach to addressing end-of-life management, including, but not limited to, subject matter expertise and resources for source storage, packaging, and transport challenges. This poster will identify the major considerations for selection, management, and secure sustainability of ionizing radiation technology, including management and replacement options, across all applications. It will also describe ORS's work to help countries, institutions, and users navigate the considerations associated with managing existing and future radioactive source usage and management and disused radioactive source disposition.

ORS provides a wide variety of resources domestically and internationally, for voluntary use, to users of radioactive materials primarily employed in medical, industrial, and research applications. To this end, ORS provides subject matter expertise, secure storage planning and disposition of disused sources, as well as targeted monetary assistance. ORS believes these considerations and resources are key in enabling a licensee to ensure both secure utilization of high-activity radioactive sources as well as successful transitions to technologically advanced machine-based radiation alternatives. ORS also provides significant support in identifying alternatives to radioactive sources for peaceful uses, and promoting the adoption and development of advanced machine-based radiation generators. Sustainable advancement of the end-user's national, security, economic, and institutional interests are key mission goals of ORS. This poster will emphasize ORS support to partners navigating the end-of-life management considerations including financial, regulatory, and logistical.

PS5 (T5.4-0862)

Optimisation of Protection – Big savings in operating costs with negligible increase in staff exposures

ANSTO is home to Australian Nuclear Science and Technical expertise. ANSTO operates one of world's most modern research reactors, OPAL; a comprehensive suite of neutron beam instruments (NBIs); **the Australian Synchrotron**, two cyclotrons, new centre for Accelerator Science. ANSTO also operates a Nuclear Medicine production facility; environmental research group; the Materials research group; a Minerals division that is a leading Uranium ore processing facility, and its own Radioactive Waste Management facility and a Radiation Instrument Calibration facility and HIFAR reactor in de-commission phase.

ANSTO has been exploring avenues in reducing business costs without compromising on worker's safety. One of the areas was suppliers of gases to all its facilities. ANSTO had realised that it would save $2 million dollars over 5 years if it changes the gas supplier. However the prospective new supplier's gas has higher concentration of Argon-40 (Ar-40), which when used in the OPAL research reactor results in activation to make Ar-41 which emits a gamma-ray with energy (E=1293KeV). This has the potential to increase stack emissions and to create higher radiation fields in occupied areas of OPAL reactor.

This poster describes the process employed by the Radiation Protection Services for Optimisation of Protection, different Radiation Protection Options considered, the Quantification of the Options and how the best Option was determined.

PS5 (T5.4-0958)

Dose response of Bacillus subtilis and Escherichia coli in different types and volumes of solutions towards gamma irradiation

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Mutagenesis of bacteria can be achieved by gamma irradiation. This technique is preferable by some as mutants induced are considered non-genetically modified organisms. Strain improvement through mutagenesis of bacteria is important to cater to the continuous demand of the bacteria bioprocessing industries for better quality strains. Mutagenesis is due to DNA damage by radiation. Also, the presence of water and thus the radiolysis of water will produce reactive oxygen species causing indirect damage to DNA. The Gamma Cell Acute Irradiation Facility at Malaysian Nuclear Agency provides irradiation services for mutation breeding. This study investigates the effect of gamma irradiation towards *Bacillus subtilis* NMBCC50025 and *Escherichia coli* DH5alpha suspended in different types and volumes of solutions. Different types of solutions including distilled water, saline solution and nutrient broth/Luria-Bertani broth were used to suspend bacteria cells. Comparisons were also performed between four different volumes (0.3 mL, 0.6 mL, 0.9 mL and 1.2 mL) of each solution. Samples were irradiated with gamma radiation from two Cs-137 sealed sources in a gamma cell. The samples were subjected to radiation doses from 0 – 1.2 kGy. The remaining viable cells after irradiation were calculated to obtain survival curves and lethal dose, 50% (LD_{50}). The amount of viable cells decreased in response to increasing irradiation doses. The LD_{50} for *Bacillus subtilis* NMBCC50025 and *Escherichia coli* DH5alpha in different volumes of solutions were in the range of 0.17-0.35 kGy and 0.013-0.043 kGy, respectively. The LD_{50}s obtained are vital to mutagenesis studies of *Bacillus subtilis* and *Escherichia coli*.

Keywords: *gamma cell, mutagenesis, survival curve*

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PS5 (T5.4-0998)

Comparison of Machine Learning-based Radioisotope Identifiers for Plastic Gamma Spectra

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Identification of radioisotopes from plastic gamma spectra is challenging because such spectra have poor energy resolutions and lack photo peaks. To overcome this weakness, many researchers have conducted radioisotope identification studies using machine learning algorithms. However, studies on machine learning based radioisotope identifiers for plastic gamma spectra are limited. In this study, machine learning-based radioisotope identifiers were implemented, and their performances were compared. Eight classes of radioisotopes consisting of combinations of ^{22}Na, ^{60}Co, and ^{137}Cs, and the background, were defined. The training set was generated by the random sampling technique based on probabilistic density functions acquired by experiments and simulations, and test set was acquired by experiments. The support vector machine (SVM), artificial neural network (ANN), and convolutional neural network (CNN) were implemented as radioisotope identifiers. The implemented identifiers were evaluated by test sets with and without gain shifts to confirm the robustness of the identifiers against the gain shift effect. Among the three machine learning-based radioisotope identifiers, prediction accuracy followed the order ANN > SVM > CNN, fluctuation on prediction accuracy followed the order CNN > ANN > SCM, while the training time followed the order SVM > ANN > CNN.

Table 1 Definition of classes for the radioisotope identification problem

Class	1	2	3	4	5	6	7	8
BKG	•	•	•	•	•	•	•	•
^{22}Na		•			•	•		•
^{60}Co			•		•		•	•
^{137}Cs				•		•	•	•

Fig. 1. Confusion matrices for radioisotope identification for combined test sets with and without gain shift using the SVM, ANN, and CNN with their best data normalization methods

Keywords: Radioisotope identification, Plastic gamma spectra, Machine learning

ACKNOWLEDGMENTS

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PS5 (T5.4-1075)

Synthesis and Scintillating Properties of Zinc Oxide Nanoparticles via Anodization

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Zinc oxide nanoparticles were manufactured using anodization method. Not only do Zinc oxide nanoparticles have wide band map, but also they have excitation binding energy 60 MeV in room temperature. So it is widely used as gas sensors, transistors, light-emitting models (LEDs) and scintillation detector. Anodization method as synthesizing metal oxide nanoparticles is green, efficient, simple, and general route.

Zinc oxide nanoparticles were manufactured by anodization in 1M KCl solution. Applying about 10V, zinc oxide nanoparticles that are less than 100nm were produced. Zinc oxide nanoparticles produced using anodization have amorphous phase. Therefore, They are required to be annealed above 450℃ to have crystalline structure. However, Zinc oxide nanoparticles are effected by surface detect and oxygen vacancy according to annealing temperature, which changes the scintillating properties.

In this study, zinc oxide nanoparticles were manufactured using anodization and the characteristics of particles were assessed according to annealing temperatures (450 ℃, 600 ℃, 800 ℃, 1000 ℃) in the air. In addition, the scintillation properties were evaluated under UV irradiation, X-ray (50 keV, tungsten target) irradiation conditions to optimize the annealing temperature for scintillating intensity. Through this data, the possibility of Zinc oxide nanoparticle as a scintillator is evaluated.

Keywords: *Zinc oxide nanoparticle, Anodization, Scintillation*

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PS5 (T5.4-1115)

Estimation of neutron production for developing an electron-based neutron source for Bragg edge imaging

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Neutron imaging is a powerful and non-destructive tool for testing materials in industrial and research applications. Compact accelerator neutron sources (CANS) are gaining interest in neutron application techniques such as Bragg edge transmission imaging. Because of their low cost and accessibilities for the users, CANS are growing worldwide. There is a pressing need for developing a neutron source in Korea for studying the structural materials. In order to develop such technologies, various technological factors need to be taken into account. The neutron production and maximization of the neutron generation, extraction and delivering to the sample position is a fundamental factor for designing a high-flux neutron source. The PHITS-3.1 [1] code has been used to determine an optimum target and moderator. Tungsten, tantalum and lead targets were considered as the target materials. The tentative calculations show that a cylindrical tungsten target with thickness of 3.5 cm and diameter of 3 cm would result in higher neutron yield. In addition, simulations were performed for different electron energies striking on the tungsten target. According to calculations, electron energy of 40 MeV was determined. Moderator material and geometry are determined in order to achieve the neutron flux of ~10^4 n/cm^2/s in the energy range of 3 to 5 meV, which are suitable for Bragg edge imaging [2], at distance of 8 m from the target. The FLUKA-2011.2x.7 [3] code was also used to calculate the neutron production yield from the designed neutron production target and the results were compared to PHITS-3.1. In this study, the preliminary results are presented for developing the neutron source for Bragg edge imaging using a small-scale electron accelerator.

***Keywords**: Neutron source, CANS, Bragg edge imaging*

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PS5 (T5.4-1147)

Development of neutron/X-ray simultaneously detectable radiation imaging module for mixed radiation inspection system

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After the September 11 attacks in the United States, the paradigms of cargo inspection system had been changed from revenuer to anti-terrorism, so importance of acquiring technique not only shape of material but also material information is growing. Because the radiation image using only X-ray can just provide material appearance, high neutron and X-ray mixed radiation cargo inspection system is needed for acquiring material information in order to detect drugs, explosives or nuclear materials. The X-ray/Neutron mixed radiation imaging system are developing at Korea National Radiation Equipment Fab. and two core technologies are developing i.e. radiation generator and imaging system including radiation sensor[1,2] and image processing. Current mixed radiation inspection system needs two separate radiation source and two imaging system. In this research, neutron/X-ray simultaneously detectable radiation sensor module was fabricated and source technology was developed for reduced fabrication cost and reduced equipment volume for portable mixed security inspection system. The results of this research would be used to neutron/X-ray mixed radiation security inspection system which have neutron/X-ray combined radiation imaging module instead of separate imaging module. Reduced fabrication cost and reduced volume are beneficial for developing portable mixed radiation security inspection system.

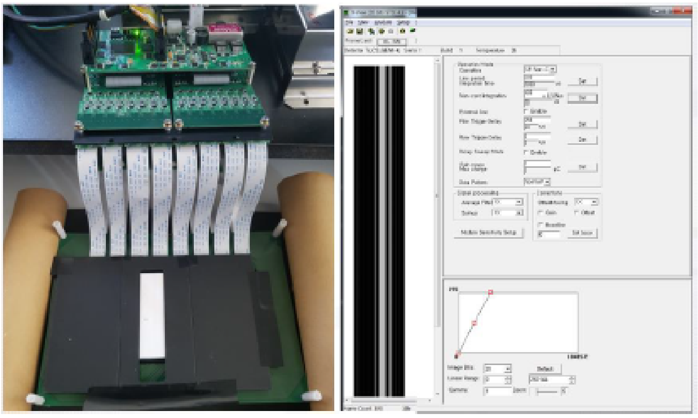

Fig. 1. Radiation image from developed simultaneously detectable radiation imaging module

Keywords: Neutron imaging, X-ray imaging, radiation inspection system

ACKNOWLEDGMENTS

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PS5 (T5.4-1148)

Design and optimization of X-ray signal processing electronics module for air cargo inspection

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We have developed signal processing electronics module including two readout electronics and a 64-channels data acquisition system for X-ray imaging systems. Readout electronics were composed of 8 X-ray sensor arrays (two $CdWO_4$ scintillators and four Si-PIN photodiodes per one detector array), 32 charge integrators, 32 sample-and-holds, a shift register, and a buffer. Data acquisition system was composed of line drivers and receivers, multiplexer, 18-bit analog-to-digital converter (ADC), and 32-kilobytes static random-access memory (SRAM). All of control signals to operate a module are implemented in the field programmable gate array (FPGA). A module operates with a single 24 V DC supply and transferred each channel data to the host computer by an Ethernet port. We built a 64-channel electronics module and evaluated its performance in terms of relative gain, offset and linearity. Tests have demonstrated that up to 16 modules can be connected. By additionally connecting 14 modules, air cargo container images were successfully acquired by irradiating an X-ray with an energy of 6-MeV.

Keywords: *X-ray imaging, signal processing, data acquisition system, air cargo inspection*

ACKNOWLEDGMENTS
This work has been carried out nuclear R&D program of the Ministry of Science and ICT of Korea (NRF No. 2017M2A2A4A05018182). It was also supported by Radiation Equipment Fabrication Center in KAERI.

PS5 (T5.4-1150)

Investigation on the Performance of Silicon Schottky Diode as Alpha-particle Detectors

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The semiconductor detectors are widely used in radiation detection because they have a small size, a high resolution in particle detection, a low operating bias voltage, and a fast time response [1]. They are commonly used for charge particle spectroscopy as well as environment monitoring. In particular, World Health Organization has already classified radon as a human carcinogen and has demonstrated a correlation between environmental radon concentration and lung cancer risk. Therefore, the development of practical detectors is necessary to measure actual dose received by indoor radon concentration exposure.

In the present work, we focused on the development of a commercially available low-cost Si detector using the simple and optimal fabrication process. We fabricated and characterized the surface barrier detectors based on high purity n-type Si substrate with a high resistivity capable of measuring Am-241 alpha particle's energy spectrum. A thin Au Schottky barrier contact with diameter of 8 mm was formed on the front side of wafer. The bottom contact of an Al layer was deposited on the back side. Further details on the fabrication of Si detectors were published in the previous work [2].

Leakage currents of in the dark are typically -12 nA/mm^2 at -12 V, which is sufficiently low enough for the counting to acquire alpha particle's energy spectra. We investigated the energy spectra of the detector at various reverse bias. The energy resolution of 2.3% FWHM was obtained for 5.48 MeV alpha particles from Am-241 which is comparable in performance with PIN diode, that are relatively more expensive. Accordingly, Si detectors of Schottky-type structure can be used in alpha spectrometric measurements. The results will be beneficial for manufacturing commercially available low-cost detectors, which can substitute advantageously expensive other detectors used for the detection and spectrometry of alpha particles.

Keywords: *Semiconductor detector, Silicon, Alpha-particle detection*

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PS5 (T5.4-1151)

Change of Leakage Current Characteristics of Si PIN Detector for X-ray depending on Post-Deposition Annealing Process

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The perception of life safety for radiation is increasing, and the use of radiation detectors, such as medical care and security, is steadily increasing. In particular, many studies are under way because semiconductor-based detectors such as silicon are capable of high accuracy and miniaturization. Si-based semiconductor detectors with silicon PIN structure has been widely used for detecting X-rays, and it requires improvement in electrical characteristics. Leakage current in semiconductor-based radiation detectors is an important factor that determines the electrical characteristics of semiconductor detectors. These leakage currents are affected by various causes, including defects and impurities present in the oxide film forming the PIN structure. This study is intended to induce changes in the characteristics of the oxide film through the post-deposition annealing process and introduce changes in the characteristics of the leakage current resulting from it.
In this study, we produced the silicon semiconductor detector for X-ray with a PIN structure scaled by 10 mm × 10 mm and conducted the post-deposition annealing process at 200-400°C in vacuum to observe the changes in leakage current characteristics. When the post-deposition annealing process is carried out, it is confirmed that there is a small leakage current of about 10 nA level under the operating voltage conditions of –70V. This shows the amount of leakage current reduced by about 50% before the post-deposition annealing process was carried out.
Based on the above results, the semiconductor detection device manufacturing process is established and we intend to manufacture high-efficiency X-ray semiconductor detection devices through more diverse experiments in the future.

Keywords: *Si, semiconductor detector, post-deposition annealing, leakage current*

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PS5 (T5.4-1152)

Result of Monte-Carlo Simulation of Material Discrimination for Air Cargo Inspection System based on 6/3 MeV X-ray and 14 MeV Neutron

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Air cargo inspection systems need the ability to quickly identify dangerous substances, such as nuclear materials, drugs and explosives from cargo containers without unpacking. Since most commercial air cargo inspection systems use high X-ray for scanning air cargo, the system is suitable for detecting materials with distinctive shapes such as weapons [1]. However, the system is difficult to distinguish drugs and explosives, which consist of elements H, C, N, and O. To improving sensitivity of material discrimination, air cargo inspection systems based on dual energy X-ray and neutron have been developed as new version of the cargo inspection system [2]. Since X-ray and neutron have different attenuation coefficients for the same materials, the inspection system based on X-ray and neutron can distinguish nuclear materials, drugs, and explosives from cargo container by comparing difference between X-ray and neutron attenuation coefficient. R-values defined as ratio of the X-ray and neutron attenuation coefficients. The R-values for 16 substances are calculated by Monte-Carlo simulation and the simulated results of 3 substances are compare with experiment result. In this paper, the calculation results of the attenuation length of 3 MeV X-ray and 14 MeV neutron for various materials using MCNP and Geant4 simulation programs.

***Keywords**: Air cargo inspection system, Security, X-ray and Neutron radiography*

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This work has been carried out under nuclear R&D program of the Ministry of Science and ICT of Korea (NRF No.2017M2A2A4A05018182). It also supported by Radiation Equipment Fabrication Center in KAERI.

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PS5 (T5.4-1170)

Assessment of External Exposure to Workers at the Geophysical Exploration Work Site by Implementing Computational Human Phantoms

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In the previous study, a logging sonde was designed for neutron-induced gamma-ray spectroscopy in a geophysics project [1]. The sonde is inserted in a borehole which is filled with water, and an Am-Be neutron source is located nearby the bottom of the sonde and shielded by a tungsten cylinder with a slit. As the sonde is utilized for underground investigations with neutron source, the neutrons from the source interact with the formation materials and water in the borehole may produce gamma-ray which could be a concern for the workers onsite with respect to the radiological protection aspect, as well as the direct exposure from the neutron flux emitted from the source.

For the calculation of effective dose, two types of phantoms are used in this study; ICRP GOLEM Voxel-type Reference Computational Phantom (VRCP) [2] and a newly developed ICRP Mesh-type Reference Computational Phantom (MRCP) [3]. Both phantoms were located at the same distance from the borehole's center and the source. The source moves from ground to the underground through the borehole. This is for estimation of effective doses as the depth of the sonde varies.

Two Monte Carlo codes, PHITS and GEANT4, are used for the calculation of doses to the workers as shown in Figure 1. The goals of this study are to estimate the whole-body effective doses by implementing computational phantoms for the workers on-site, and comparing the results from PHITS and GEANT4. The effective dose evaluation for the workers could provide a basis for setting up the radiological protection plan for the geophysical exploration procedure.

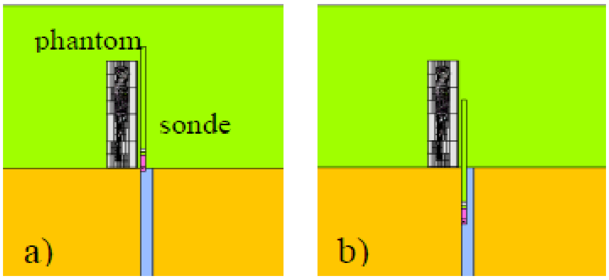

Figure 1. The Monte Carlo simulation geometry with MRCP. (a) The source is placed at the ground. (b) The source is placed at a depth of 1 m underground.

Keywords: External exposure, Dose assessment, Computational Phantom, Monte Carlo simulation

ACKNOWLEDGMENTS

This work was supported by Korea Environment Industry & Technology Institute (KEITI) through Subsurface Environment Management (SEM) Project, funded by Korea Ministry of Environment (MOE) (2018002440004).

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PS5 (T5.4-1191)

Analysis of the Status of the Facilities Used for Radiographic Testing in KOREA

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The industrial radiographic testing workplaces (called by 'temporary job site') were divided into use facility, shop, and field. Technical standards for use facility as amended in 2019 include access control devices, radiation detection devices, remote controllers, etc. Therefore, the facilities used for radiographic testing have been inspected to confirm whether the technical standards were applied. The objective of this study was to analyze the status of the facilities used for radiographic testing in order to improve safety of radiographic workplaces. We analyzed the inspection results of facilities used for radiographic testing, the status of notified radioisotope, radiation generating device (RG), and radiation safety officers by workplaces. Based on the inspection results, the facilities used for radiographic testing were analyzed for the size of the facility, the thickness of the shielding wall, and the presence of ceiling shielding. The status of notified sealed radioisotopes, radiation generating devices, radiation safety officer, and occupational radiation exposure by workplaces was analyzed through the analysis of workplace for radiographic testing declaration in 2019. The size distribution of the facilities used for radiographic testing in KOREA was (width) 1,800 to 65,000 mm, (length) 1,250 to 26,200 mm, (height) 1,850 to 21,000 mm. Also, the average size of the facilities used for radiographic testing was (width) 10,000 mm × (length) 5,000 mm × (height) 4,000 mm, with the thickness of the shielding wall of 1,000 mm and the ceiling shielding of 400 mm. The proportion of notified sealed radioisotope and radiation generating device by 616 workplaces was 53% (Ir-192) 28% (radiation generating device), 10% (Se-75), and 9% (Co-60), respectively (see Fig.1). There were designated 392 radiation safety officers to the 616 use facilities, and 36% of the radiation safety officers were assigned to one or more workplaces. Facilities used for radiographic testing by region are mainly located near shipbuilding and chemical plant. The average annual radiation exposure of 4,343 radiation workers at the use facility was 2.89 mSv. We analyzed the status of facilities used for radiographic testing in order to improve safety of radiographic workplaces in 2019. The results of this study will be used to amend the safety control of mobile use in the facilities used for radiographic testing in KOREA.

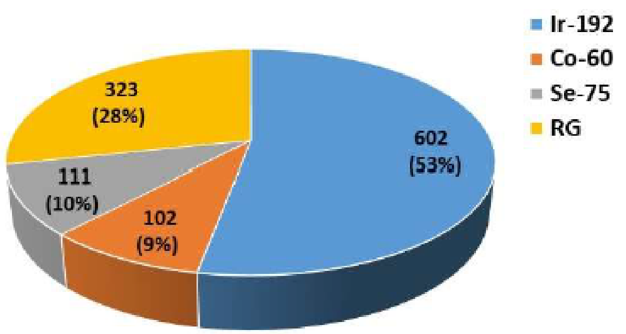

Fig. 1. The status of notified sealed radioisotope, radiation generating device in 2019

Keywords: Use Facility, Radiographic testing, Temporary job site

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PS5 (T5.4-1217)

Beamline design and beam transport calculation for µSR facility in RAON considering the radiation shielding design

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The heavy-ion accelerator complex, RAON, is currently under construction in Korea by the Rare Isotope Science Project (RISP). RAON will provide a variety of RI beams and stable beams to support a wide range of research such as nuclear science, biomedical science and applied science [1]. A µSR facility will be installed as one of the main facilities for material science. The µSR facility will use surface muons generated from pions decaying at rest which are products of the interaction between 600 MeV proton and a muon production target. In this paper, the design of the muon beamline using optics calculation and the beam transport calculation using ray-tracing are described. Beam optics calculation was carried out to determine the positions and the specifications of electromagnetic components by using GICOSY code [2]. The calculation was done up to the fifth-order taking into account large emittance of the muon beam, and the fringe field effect was applied for every component. The configuration of the beamline was determined considering the facility's shielding design based on the result of the calculation; It consists of 2 solenoids, 2 dipoles with a deflection angle of ±70 degrees, 9 quadrupoles and a Wien filter, and the total length is 18 m. In order to verify the configuration, a ray-tracing calculation was carried out using G4beamline code [3]. The initial distributions at the target of position, angular divergences and momentum of muons derived from Geant4 simulation were applied. The fringe field information was derived from the electromagnets designed by using OPERA-3D code [4]. In addition, rotation of muon spin angle and separation of other particles through the Wien filter were estimated. The flux of muons entering the first solenoid was 4.2×10^7 μ^+/s (solid angle of 107 msr) and the flux at the 1.5×1.5 cm^2 sample was estimated to be 5.2×10^5 μ^+/s (transmission of 1.23%) at a proton current of 60 µA.

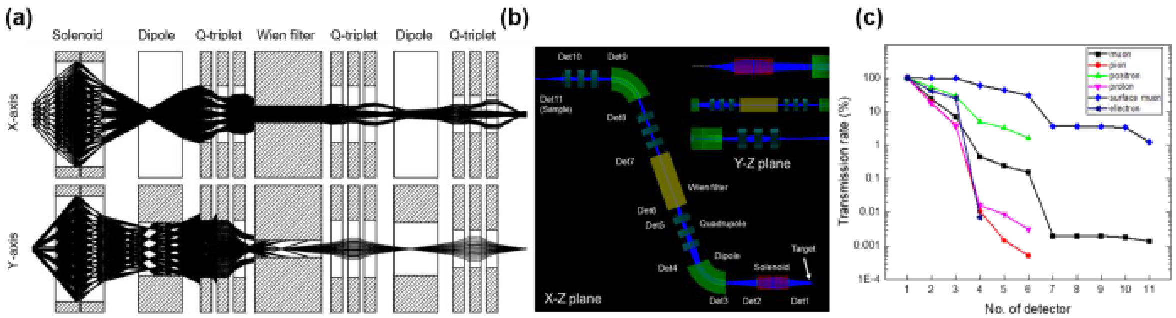

Fig. 1. (a) Results of optics calculation (b) Results of ray-tracing calculation (c) Transmission rate by particle

Keywords: RAON, surface muon, beamline

ACKNOWLEDGMENTS
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PS5 (T5.4-1217)

A Simulation Study on Effect of Collimator Design on Imaging Performance of Large-area Hybrid Gamma Imaging System

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Hybrid gamma imaging features a broad dynamic energy range, which is desirable for fields where imaging several isotopes is required such as industrial survey and environmental remediation. By fusing data from mechanical and electronic collimation imaging (i. e. coded aperture and Compton imaging), inherent energy dependencies of two imaging methods are compensated in hybrid imaging. When composing hybrid imaging system by combining coded aperture collimator and Compton camera, the collimator design affects performance of both coded aperture and Compton imaging, and the balance of such trade-off is highly dependent on system configurations such as material and size of the mask and the detectors. Hence, the effect of the collimator design on imaging performance of the hybrid imaging system should be studied specifically to the detector. In the present study, the effect of the collimator design on the performance of the large-area hybrid gamma imaging system was surveyed with Geant4 Monte Carlo simulations. The large-area hybrid gamma imaging system, composed of two scintillation detectors and a coded aperture collimator mask, inherits its detectors from Large-area Compton camera (LACC) [1]. The tungsten mask is patterned as modified uniformly redundant array and placed at 6 cm in front of the front detector, considering field of view. The system was modeled in Gean4 Monte Carlo simulation. It was assumed that a point source was placed at 3 m from the system and was measured for 1 minute. Background radiations were also simulated. Imaging resolution and imaging sensitivity were surveyed with several conditions with various energies (100–2000keV) and mask thicknesses (2–10 mm). Minimum detectable activity (MDA) was considered as an index of imaging sensitivity. The MDA was derived using receiver operator characteristic analysis. The MDA of the hybrid imaging system at low energy (<200 keV) and high energy (>1000 keV) was less dependent on the mask thickness. On the other hand, at intermediate energy (400–800 keV), 2-mm mask showed higher MDA than others. For the mask with 4–10 mm thickness, it showed similar MDA on this energy range. Imaging resolution also showed a similar trend to the MDA. A significant effect of mask thickness on imaging resolution was observed on intermediate energy range (400–800 keV). Imaging resolution was degraded as the mask thickness decreased. The findings from the present study will be utilized to design the mask of the large-area hybrid gamma imaging system.

Keywords: *Large-area hybrid gamma imaging system, Minimum detectable activity, Monte Carlo simulation*

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PS5 (T5.4-1217)

Experimental feasibility study of LACC-based Compton Camera Computed Tomography

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For imaging the location of radiation source or hot-spot contamination in a radioactive waste drum, we are planning to use the Large-area Compton Camera (LACC) [1] with a statistical image reconstruction algorithm (ML-EM). The use of the ML-EM algorithm, however, requires the distribution of linear attenuation coefficient (i.e., attenuation map) for the waste drum. In the present study, we proposed and experimentally demonstrated a new imaging method called Compton Camera Computed Tomography (C^3T), which uses a Compton camera (e.g., LACC) as the detector to obtain the attenuation map. The LACC-based C^3T consists of a ^{137}Cs gamma-ray beam source, a drum rotation system, and the LACC. In the present study, a very low activity (= 250 μCi) ^{137}Cs point source was used as the gamma-ray beam source. The source was located at 2 m distant from the LACC. To test the feasibility of the Compton CT experimentally, a lead brick of 10 cm (L)× 5 cm (W) × 20 cm (H) was placed at the center of the drum rotation system, and the LACC measured the gamma-rays from the ^{137}Cs source while rotating the lead brick using the drum rotation system. The distance between the center of the lead brick and the LACC was 40 cm. The lead brick was rotated through 360° at intervals of 5°. The measurement time was set to very long (= 1 hour per projection) due to the low activity of the source. The attenuation map was then reconstructed by the filtered back projection with Ram-Lak filter. Image reconstruction was done considering a fan beam source. Fig. 1(a) shows the 2D reconstructed attenuation map image for the lead brick with the red square line which indicates the true shape of the brick. The reconstructed image well matches the true shape of the brick. Fig. 1(b) and Fig. 1(c) show 1D profiles for the horizontal (X-axis) and vertical (Y-axis) directions. The results show that the Compton CT can reproduce the value of the true linear attenuation coefficient. The reconstruction error was calculated to be 0.59. Some artifacts were observed in the reconstructed image due to an experimental alignment error during the long measurement time from the low-intensity source. The image quality is expected to be significantly improved by using a higher activity beam source. In the near future, we will carry out various experiments to obtain the attenuation maps for various materials and objects.

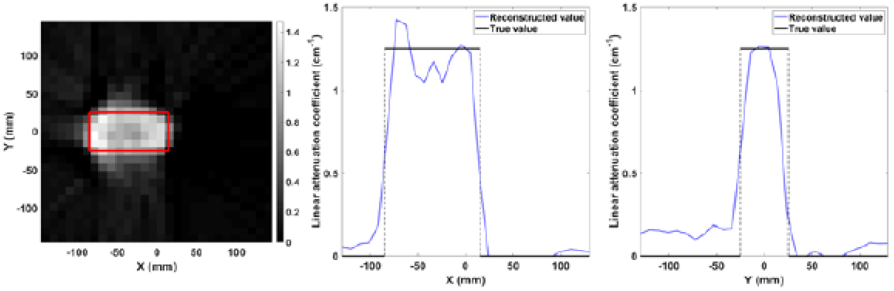

Fig. 1. Reconstructed attenuation map for lead brick (a) and 1-D profile image for X-axis (b) and Y-axis (c)

***Keywords**: Compton camera, computed tomography, attenuation map, image reconstruction*

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PS5 (T5.4-1235)

A Simulation Study on Effect of Collimator Design on Imaging Performance of Large-area Hybrid Gamma Imaging System

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Hybrid gamma imaging features a broad dynamic energy range, which is desirable for fields where imaging several isotopes is required such as industrial survey and environmental remediation. By fusing data from mechanical and electronic collimation imaging (i. e. coded aperture and Compton imaging), inherent energy dependencies of two imaging methods are compensated in hybrid imaging. When composing hybrid imaging system by combining coded aperture collimator and Compton camera, the collimator design affects performance of both coded aperture and Compton imaging, and the balance of such trade-off is highly dependent on system configurations such as material and size of the mask and the detectors. Hence, the effect of the collimator design on imaging performance of the hybrid imaging system should be studied specifically to the detector. In the present study, the effect of the collimator design on the performance of the large-area hybrid gamma imaging system was surveyed with Geant4 Monte Carlo simulations. The large-area hybrid gamma imaging system, composed of two scintillation detectors and a coded aperture collimator mask, inherits its detectors from Large-area Compton camera (LACC) [1]. The tungsten mask is patterned as modified uniformly redundant array and placed at 6 cm in front of the front detector, considering field of view. The system was modeled in Gean4 Monte Carlo simulation. It was assumed that a point source was placed at 3 m from the system and was measured for 1 minute. Background radiations were also simulated. Imaging resolution and imaging sensitivity were surveyed with several conditions with various energies (100–2000keV) and mask thicknesses (2–10 mm). Minimum detectable activity (MDA) was considered as an index of imaging sensitivity. The MDA was derived using receiver operator characteristic analysis. The MDA of the hybrid imaging system at low energy (<200 keV) and high energy (>1000 keV) was less dependent on the mask thickness. On the other hand, at intermediate energy (400–800 keV), 2-mm mask showed higher MDA than others. For the mask with 4–10 mm thickness, it showed similar MDA on this energy range. Imaging resolution also showed a similar trend to the MDA. A significant effect of mask thickness on imaging resolution was observed on intermediate energy range (400–800 keV). Imaging resolution was degraded as the mask thickness decreased. The findings from the present study will be utilized to design the mask of the large-area hybrid gamma imaging system.

Keywords: *Large-area hybrid gamma imaging system, Minimum detectable activity, Monte Carlo simulation*

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PS5 (T5.5-1016)

Seasonal Effects on Flame Temperature in Open Pool Fires

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Dual-purpose casks are used for the storage and transport of spent nuclear fuel assemblies. They must therefore satisfy the requirements prescribed in the Korea Nuclear Safety Security Commission Act 2017-56, the IAEA Safety Standard Series No. SSR-6, and US 10 CFR Part 71. These regulatory guidelines classify a dual-purpose cask as a Type B package and state that a Type B package must be able to withstand a temperature of 800 °C for a period of 30 min. In general, it is found that the flame temperature is affected by radiation from a cooler surface. Therefore, open-pool fire tests were conducted in summer and winter to estimate the effect of season on flame temperatures using a one-sixth slice of a real cask. In the open-pool fire test conducted in the winter, the average flame temperature was 834 °C and the neutron shield reached a maximum temperature of 183 °C. This indicates that the dual-purpose cask was properly insulated from the heat of the flames. The temperature rise of the basket during the fire test was 29 °C. In the open-pool fire test conducted in the summer, the average flame temperature was 851 °C. The neutron shield reached a maximum temperature of 151 °C. The temperature rise of the basket during the fire test was 35 °C. As a result, it can be seen that the flame temperature measured was higher in the summer than in the winter. Therefore, it would be desirable to perform open-pool fire tests in the summer. In addition, the thermal integrity of the dual-purpose cask was maintained and the cask is judged to be sufficiently safe under thermal condition of 800 °C.

Keywords: *Dual-purpose cask, Type B package, Open-pool fire test, Flame temperature, Thermal integrity*

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PS5 (T5.6-0388)

On the Roles of a Radiation Protection Expert - Experiences from Decommissioning A Hot Cells Facility

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The purpose of this abstract is to share the main experiences gained from the radiation protection of the decommissioning of the Danish hot cells facility at Risø. The project framework is discussed.

The framework of the project is setup by Danish Decommissioning (DD) and the Danish authorities. The setup is an adapted stage gate model. Before a major decommissioning project is allowed to start by the Danish authorities, a project description has to be made. This document serves as the project charter and must include a description of the project in terms of milestones and the goals and overall activities of the milestones. This adapted stage gate model differs from typical commercial projects in the fashion, which in typical projects the results achieved in the given stage is evaluated and the possibility of completing the project successfully is assessed, whereas the plan and risk assessment of each stage is evaluated in the adapted stage gate model. This adapted stage gate model is a very good model to ensure that possible doses have been optimized. The adapted stage gate model is very well suited for managing decommissioning projects.

Although having an advisory and supervising role the health physicist of the hot cell project participates actively in all aspects of the planning in the project and in the execution of the operations. In a day to day practical approach ALARA is not a convenient A Low Aa possible. The minimizing approach helps to ensure an awareness of safety.

Essentially the most effective and important applied radiation protection method arises from the workers doing the actual work. Their safety culture and way of working is a paramount parameter for a successful radiation protection scheme. The workers need to understand why and how they can reduce external doses and risk of internal doses. That knowledge must reflect in the way they are working. The presence of the Health assistant is also very important.

The second most effective method of applied radiation protection is the engineering team. Simply by, acknowledge the importance of continually thinking dose and risk reduction into both planning and execution. In the hot cell project, an active role of the health physicist in the project helped achieving this.

Mock ups are an effective tool in radiation protection.

Besides the usual supervision and advisory role of the health physicist there is the role of a facilitator of radiation protection, helping all aspects of the decommissioning project to think and live radiation protection. Effective radiation protection is not due to the health physicist but due to the project engineers and the workers, the health physicist has to be able to facilitate this.

Keywords: *Facilitator role, safety culture, practical experience*

PS5 (T5.6-0799)

Comparison of Preliminary $DCGL_w$s between Rancho Seco and Kori Unit-1 for surface soil according to the composition of the soil layer

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Kori Unit-1, the first PWR (Pressurized Water Reactor) plant in South Korea, began its commercial operation in 1978 and was permanently shut down on June 19, 2017. In order to release nuclear facility site after decommissioning, licensee should demonstrate that decommissioning site comply with release criteria (10 mrem/y in South Korea). The release criteria were calculated as the Derived Concentration Guideline Levels (DCGL) using RESRAD-ONSITE and inputs for calculation include information of contamination zone, unsaturated zone etc. In this study, $DCGL_w$s were calculated according to the composition of soil layer. Although the characteristic of Kori Unit-1 and Rancho Seco are not similar, Kori Unit-1 was analyzed based on Rancho Seco's decommissioning data in terms of radiological characterization of contaminated site. Rancho Seco has four unsaturated zones, which consisted of silt, sand, silt, and sand, and has one saturated zone, which consisted of sand-stone. The Kori Unit-1 was assumed to have three unsaturated zones, which consisted of sandy loam, loam, and loam, and has one saturated zone which consisted of sand-stone. Rancho Seco termination plan's appendix 6-A was used to refer the other inputs parameter. As a result, ^{14}C has largest differences between the $DCGL_w$s of Rancho Seco and Kori Unit-1 and ^{137}Cs has smallest differences as seen in Table 1. The mean distribution coefficients for ^{14}C, ^{63}Ni and ^{137}Cs are 2.40, 6.05 and 6.10, respectively. Although the distribution coefficient of ^{63}Ni and ^{137}Cs are most similar to each other compared to other nuclides, differences between the $DCGL_w$s of Rancho Seco and Kori Unit-1 were considerably different. Therefore, when $DCGL_w$s for actual decommissioning site were calculated, the composition of soil layer should be properly considered.

Table 1. $DCGL_w$ for Surface Soil (based on 25mrem/y)

	Rancho Seco [pCi/g]	Kori Unit-1 [pCi/g]	Relative Difference [%]	Mean Distribution Coefficients [cm³/g]
^{14}C	8.330E+06	2.866E+03	99.9	2.40
^{60}Co	1.260E+01	1.109E+01	11.98	5.46
^{63}Ni	1.520E+07	4.028E+06	73.5	6.05
^{90}Sr	6.490E+03	2.969E+03	54.25	3.45
^{134}Cs	2.240E+01	1.997E+01	10.84	6.10
^{137}Cs	5.280E+01	4.748E+01	10.07	6.10

Keywords: Surface Soil, $DCGL_W$, RESRAD

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PS5 (T5.6-0869)

Details of Uncertainty Analysis on Decommissioning Cost of Nuclear Power Plant

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In Korea, as a result of the permanent shutdown of Kori Unit 1 and subsequent Wolsong Unit 1 over the last few years, much attention has been given to the decommissioning of nuclear facilities as part of the new government's nuclear policy. Decommissioning is the last stage in back-end cycle of nuclear power plant and could be defined as all of the administrative and technical actions associated with early planning for cessation of operations through termination of all licenses and release of the site from nuclear regulatory control [1]. In order to achieve and complete safe and economical decommissioning projects, decommissioning cost estimation has been recommended by regulatory authorities. And also, decommissioning cost is taken into consideration by corresponding utility and engineering organizations.

However, most of decommissioning costs evaluated at the stage of project planning have a variety of uncertainties that cause inaccuracy in calculating appropriate cost. Uncertainties are broadly classified into three kinds of categories, routine variability, insufficient knowledge and risks [1]. In the majority of cases including Korea, uncertainties until the end of decommissioning are critical to cost calculation because existence of uncertainties, especially qualitative uncertainties, makes it impossible to guarantee the preciseness of decommissioning costs continuously. Nevertheless, financial resources such as provisions for decommissioning is going to be provided based on anticipated costs in accordance with regulations.

Accordingly, in this paper, analysis on uncertainty and development on risk mitigation methods for the decommissioning costs of nuclear power plants was performed and is being investigated in the future. The purpose of this study is to quantify or anticipate qualitative factors among the uncertainty of cost estimation.

Parameters in Korean decommissioning cost related or similar to uncertainty were examined and each characteristics was distinguished as depicted in Table 1. Contingency is often used as a bundle of cost including all unclear factors in the past and can be regarded as estimating uncertainty within the defined decommissioning. Allowance is contained within dismantling and decontamination cost as work difficulty factors including radiological environments in workplace. Risk for decommissioning cost could be referred as out-of-scope uncertainty, which can be divided to funded and unfunded according to availability of financial resource preparation.

Table 1. Main factors and impacts of risk on decommissioning cost

Uncertainty	Definition / Characteristics
Contingency	Bundle of cost for all unclear factors in Korea (Example : Cost for waste disposal)
Allowance	Provision for known activities but not exactly presented (Included in base cost)
Estimating Uncertainty	Provision about in-scope uncertainty (Regarded as contingency)
Funded Risk	Out-of-scope uncertainty be prepared or provided by project management organization
Unfunded Risk	Out-of-scope uncertainty not be provided within the decommissioning project

Keywords: Decommissioning Cost Estimation, Uncertainty Analysis, Contingency in Decommissioning Cost

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PS5 (T5.6-0870)

A Study on Decommissioning Site Restoration Technology for NPP in Korea

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Decommissioning defined by IAEA means the administrative and technical actions taken to allow the removal of some or all of the regulatory controls from a facility [1]. In Korean Nuclear Safety Acts, decommissioning refers to all the activities for regulatory exclusion by demolishing buildings and site or removing radioactive contamination. In order to accomplish the definition or objectives, site restoration and remediation should be carried out as the last phase of decommissioning project. Site restoration is going to be conducted for treating and utilizing the released site to be compliance with 0.1 mSv/yr for individual effective dose rate by residual radioactivity for 1,000 years from the decommissioning site release as the regulatory criteria noticed by Nuclear Safety and Security Commission [2].

Meanwhile, as specified in Korean nuclear regulations, disposal unit cost per 200 L drum is intimately associated with the existence of radioactivity in surface water, groundwater and soil waste. If the radioactive waste has specific radioactivity higher than clearance criteria. The disposal unit cost of radioactive waste is 14.36 million Korean Won not related with level of radioactive waste such as intermediate, low and very low level. For lessening the burden caused by disposal unit cost of radioactive waste, utilizing aggressive and challenging site restoration technologies could be contributed to economic decommissioning in Korea.

A number of site restoration and remediation technologies are introduced theoretically in EURSSEM and corresponding reports [3, 4]. Because of high necessity to decontamination and volume reduction of radioactive water and soil waste, literature survey and case study of site restoration and decontamination technologies are progressed.

Site restoration technology could be categorized by contamination targets. As shown in Table 1, research and development of filtration and reverse osmosis hybrid technology for surface water and groundwater, combination technology of radioactivity measurement and separation after washing for soil are going to be investigated in this study. Finally, some technologies will be developed in the near future.

Table 1. Classification and R&D Progress of Site Restoration Technology

Classification	Selected Technology	R&D Progress
Surface Water & Groundwater Remediation	Filtration & Reverse Osmosis Hybrid Process	Design and Construction
Soil Remediation	Soil Washing, Activity Measurement & Separation	Design and Construction

Keywords: Site Restoration, Surface Water/Groundwater/Soil Remediation, Soil Decontamination

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PS5 (T5.6-0886)

Performance evaluation of portable scanning system for radiologically contaminated sites

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A portable scanning system for radiologically contaminated sites is developed and evaluated to analyze gross alpha-beta and gamma radioactivity, as well as gamma dose rate by evaluation of equivalent dose factors. As shown in Fig. 1 [1], PC server obtains signals from separate gross alpha-beta arrays and gamma-ray spectrometer. The gross alpha-beta arrays comprise ZnS(Ag) and PVT phoswich detector, and the gamma-ray spectrometer is composed of Compton-suppressed anti-coincidence circuit between NaI(Tl) and polyvinyltoluene (PVT) scintillators.

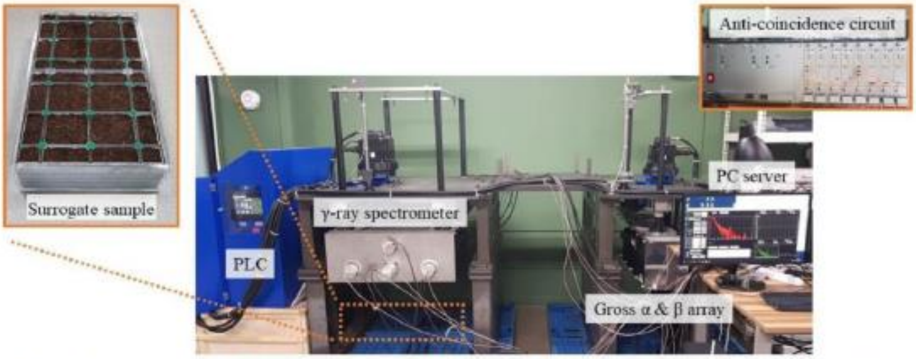

Fig. 1. Experimental setup for validation of the gamma-ray spectrometer [1].

For performance evaluation of the system, energy spectra are obtained for several important radionuclides (e.g., ^{60}Co and ^{137}Cs) in soil, and key indicating parameters are calculated, such as detection efficiency, signal-to-background-ratio, and minimum detectable activity. Especially, surrogate samples are made in laboratory by using soild check sources and agricultural soil. Monte Carlo N-Particles 6 (MCNP6) code [2] is used as a compensation tool to simulate the gross alpha-beta arrays and gamma spectrometer in homogeneous contamination conditions. Then, the system is tested in actual field for comparison against laboratory-scale measurement. Even in harsh condition with conservative assumption (e.g., without Compton suppression, radionuclides buried at depth of 0.05 m…), the results represent that our system can measure sufficiently low-level radioactivity as nuclear decommissioning sites (~15 kBq/m^2 for ^{60}Co), aiming at green field (residential farmer scenario in Korea, that is, 0.1 mSv/yr), with scan coverage of 10,000 m^2 within ~ 30 min, thanks to sufficient efficiency (~2% and ~0.5% for measurement of ^{137}Cs and ^{60}Co at 0.1 m).

Keywords: Scan coverage, Signal-to-background ratio, Minimum detectable concentration

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PS5 (T5.6-0892)

Sensitivity evaluation of Scan MDC for various detectors for final status survey in decommissioning site

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The release of site has usually been considered as the final step in a sequence of decommissioning of the nuclear facilities. Site characterization in the final status survey after decommissioning is an important step for planning and implementation of site remediation and release [1]. Radiological survey and evaluation technology are required to ensure the reliability of the results, and the process must be easily applied during field measurements. Scan survey is performed to identify the presence of any locations of elevated contamination. The probability of detecting residual radioactivity in the field is affected not only the efficiency of the survey instruments but also the survey conditions [2]. The minimum detectable concentration of scan scurvy depends on the intrinsic characterization survey instruments and distribution of contamination. In this study, we report a scan MDC for various sizes of NaI detector and $LaBr_3$(Ce) detector and evaluation methods.

The evaluation result of a scan MDC 2”× 2” NaI(Tl) detector using MCNP code was 4% of relative error by compared with the results of NUREG-1507. The sensitivity of survey conditions such as a scan speed and detector height was evaluated for the scan MDC for field survey.

Table 1. NaI and LaBr3 detector scan MDC for ^{137}Cs contamination

	2”× 2” NaI(Tl)	3”× 3” NaI(Tl)	2”× 2” LaBr3
Relative efficiency(%)	100	233	108
MDCR Surveyor(cpm)	1512	2308	1571
Scan MDC(Bq/g)	0.25	0.16	0.26

Keywords: *Decommissioning site, final status survey, scan MDC, gamma-ray survey*

ACKNOWLEDGMENTS

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PS5 (T5.6-0928)

Groundwater Monitoring Experiences in the U.S during Decommissioning of Nuclear Power Plant

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Kori Unit 1 is currently in transition period of at least 5 years as it has been permanently shut down in June 2017 and is expected to start decommissioning in 2022. This will lead Korea to stand in the ranks of countries with experience in construction and operation of nuclear facilities as well as decommissioning. An important part of decommissioning would be to ensure the safety of workers and residents from radiation contamination and exposure, and to reduce costs through efficient preparation for the decommissioning. However, in the U.S. decommissioning experience, one of the areas of concern is that inadvertent contamination of radionuclides were spread into the site during operations period, which caused groundwater contamination and, consequently, incurred considerable human resources and costs for treatments at the time of site remediation. Therefore, it will be necessary to highlight lessons learned from oversees experience in establishing groundwater monitoring programs to prevent costly groundwater contamination.

The purpose of this study is to review the lessons learned from the groundwater contamination and characterization experience of Yankee Nuclear Power Station (YNPS) and Connecticut Yankee (CY) nuclear power plant in the U.S., and to find implications for the decommissioning of Kori Unit 1. The YNPS was a 185 MWe pressurized water reactor, operated from 1961 to 1991, and License Termination Plan (LTP) was approved in 2006. The groundwater contamination was suspected in the early 1960s in which a large quantity of contaminated water from the pit of ion exchanger leaked through a defect of a concrete wall. In the Historical Site Assessment (HSA), it was confirmed that contamination was from the leakage of an underground pipe of the liquid waste treatment system and Spent Fuel Pool (SFP). On the other hand, the CY plant was a 619 MWe pressurized water reactor, operated from 1968 to 1996 and decommissioning was completed in 2007. The CY also found contamination during HSA and characterization, with many of radionuclides contaminating bedrocks, and about 20 radionuclides were analyzed and investigated the areas affected. After remediation of groundwater and soil, an additional 18 months of sample analysis recommended by State of Connecticut Department of Environmental Protection was performed, and the license termination was approved after confirming the stability trend of contamination. Table 1 summarizes the considerations for groundwater monitoring based on experience from the YNPS and the CY.

Table 1. Lessons learned from the groundwater monitoring program in the U.S. [1]

YNPS	CY
✓ An early development of comprehensive Site Conceptual Model (SCM) ✓ Efforts to identify data gaps and verify the model ✓ An extended investigation requires for about 3 years ✓ The decisions to install additional wells and extended investigation were made with stakeholders ✓ The investigations is likely to expedite the long term monitoring ✓ The YNPS investigations required a significant expenditure in resources	✓ The details of the local hydrogeological features must be sufficiently described ✓ Plant managers should engage all stakeholders early ✓ Investigation and monitoring should begin early ✓ The primary radionuclides of concern with regard to groundwater impact are important ✓ Groundwater investigations should be undertaken with an iterative approach ✓ Investigations should consider the locations and depths dictated by and understanding of the site

Keywords: Decommissioning, Groundwater Monitoring, Characterization

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PS5 (T5.6-0929)

A Review on the Site Release Limit of Groundwater from the Decommissioning of Nuclear Power Plant

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At the license termination, the level of radionuclide concentrations in the media and buildings remaining on the site of the nuclear installation should be less than or equal to those of release limits. In general, the media such as soil, groundwater or buildings within the site may be considered for released in the evaluation of potential reuse scenarios. Particularly in the case of groundwater, an appropriate allotment of dose should be well distinguished when deriving soil and groundwater Derived Concentration Guideline Levels (DCGLs) because exposure pathways that related to groundwater are included in deriving soil DCGLs. Therefore, this study intends to review the radionuclides of interest, release limits, and Environment Protection Agency (EPA) Minimum Concentration Levels (MCLs) from the case of facilities that have been decommissioned.

In the License Termination Plan (LTP) of the Yankee Nuclear Power Station (YNPS), H-3 (tritium) was considered as a major radionuclide of groundwater. They calculated the dose contribution form groundwater at the EPA MCLs and found that the dose due to H-3 was estimated to be 7.7 µSv/yr. A case of Connecticut Yankee (CY) plant, three types of contamination of soil, groundwater and subsurface concrete are considered for dose calculation due to the present of significant groundwater contamination. It should be note that the total dose was taken into account in all exposure pathways that can be received from the site's residual radioactivity, where groundwater contamination represents the dose contribution by the current residual radioactivity in groundwater. In consideration of site release limits, DCGLs reflecting site characteristics are derived, but U.S. nuclear power plants have established EPA MCLs as a groundwater release limit in accordance with the EPA/NRC memorandum of understanding. Therefore, CY considered both of DCGLs and MCLs as groundwater release criteria, and no radionuclides that exceed 0.01 mSv/yr were detected in groundwater, comparing a Total Effective Dose Equivalent (TEDE) dose at the MCL concentration [1].

Table 1. Comparison of limits for some radionuclides in groundwater [1, 2]

Radionuclide	WHO (Bq/L)	MCLs (Bq/L)	DCGLs (Bq/L)	Radionuclide	WHO (Bq/L)	MCLs (Bq/L)	DCGLs (Bq/L)
H-3	1.0E+04	740	2.4E+04	Ni-63	1,000	1.9	120
C-14	100	74	330	Sr-90	10	0.3	9.3
Fe-55	1,000	74	2.4E+03	Cs-134	10	3	13
Co-60	100	3.7	42	Cs-137	10	7.4	16

Unlike DCGLs applied to specific sites, some nuclear power plants may refer to the values proposed by the World Health Organization (WHO) as a guidance for the site release levels for groundwater. The radionuclide concentration levels presented by DCGLs, MCLs and WHO correspond to dose of 0.25 mSv/yr, 0.04 mSv/yr and 0.1 mSv/yr, respectively. Therefore, as shown in Table 1, it can be seen that for H-3 as an example, the concentration of DCGLs derived from 0.25 mSv/yr criterion is 2.4 times higher than the concentration derived from 0.1 mSv/yr criterion of WHO. Although there may some differences depending on radionuclides, overall, the release limits increase in the order of MCLs, WHO, and DCGLs. The standard for the release limit of groundwater can be referred with an understanding of the plants that have experienced decommissioning.

Keywords: Decommissioning, DCGLs, MCLs, Groundwater

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PS5 (T5.6-0936)

Internal Dose Calculation Using High-resolution Measurement of Aerosols from Plasma Arc Torch Cuts on Stainless Steel

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Cutting large radioactive structures into acceptable size for waste disposal is an essential process in decommissioning nuclear power plants. For cutting metallic waste, a plasma arc torch is commonly used because of fast cutting speed. However, plasma arc cutting has the disadvantage of generating a large number of radioactive aerosols [1]. For workers' radiological safety, it is essential to find out the physical and chemical properties of aerosols from cutting processes such as chemical form, radionuclides component ratio, and size distribution of aerosol [2]. To investigate the characteristics of aerosols from plasma arc cut on stainless steel in a different cutting condition, a measurement system was designed to control cutting speed and possible to isokinetic sampling [3]. The aerosol measurement system was composed of a cutting chamber (ca. 0.86 m^3), an x-stage automatic cutting system using servo motor, the plasma arc torch (Powermax125, Hypertherm), and high-resolution electrical low-pressure impactor (HR-ELPI+, Dekati). 10 mm thickness stainless steel 304 plates were cut in conditions: Cutting length set as 3 cm, cutting speed set as 10 mm/sec, and plasma arc current set as 75 A. IMBA® internal dosimetry software was used to calculate the internal dose in four different ways: 1 and 5 μm as activity median aerodynamic diameter(AMAD), calculated AMAD from high-resolution number distribution, and calculated AMAD of ^{55}Fe from ICP-OES analysis (Table 1). The high-resolution number distribution of aerosol was converted into MMAD (mass median aerodynamic diameter) using the Hatch-Coate equation. Because the isotope distribution was independent of aerosol size [4], it was assumed that the measured MMAD values could be used as AMAD value. The effective dose coefficient using 5 μm showed the lowest value among the other calculated effective dose coefficients. The internal dose by radioactive aerosol inhalation without considering the characteristic of aerosol can be assessed lower than the actual internal dose. Therefore, we concluded that the potential radiation risk to workers may have been underestimated in the process of decommissioning nuclear power plants.

Table 1. Effective dose coefficient calculated by using IMBA® internal dosimetry software

Dose calculation input (^{55}Fe)	Effective dose coefficient [Sv/Bq]
AMAD (1 μm) (ICRP Reference for public)	3.70E-10
AMAD (5 μm) (ICRP Reference for worker)	3.30E-10
Calculated AMAD from High-resolution number distribution	4.12E-10
Calculated AMAD from particle size distribution of iron	6.20E-10

Keywords: Decommissioning, Radioactive aerosol, Waste management

ACKNOWLEDGMENTS

This work was supported by the National Research Foundation of Korea(NRF) funded by the Korea government (MSIT: Ministry of Science and ICT) (Grant No. NRF-2017M2A8A4018596).

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PS5 (T5.6-0994)

Development of cost evaluation methodology for selection of optimal radioactive waste treatment process

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The establishment of decommissioning plan for the nuclear facilities is an important factor for successful decommissioning projects. In particular, it is important to establish a waste treatment process plan in consideration of waste reduction and cost.

Therefore, this paper describes a methodology for evaluating cost, which is one of the factors considered for the treatment of various and massive wastes generated during the decommissioning process, and established unit cost data for cost evaluation. In addition, it was verified based on the only experience data in Korea. The unit cost data was established based on domestic and overseas decommissioning experience data, and the cost evaluation method for each waste treatment process was developed by referring to the general cost calculation methodology. The cost items were categorized into labor cost, equipment/material cost, operating cost, and nuclide analysis cost. The cost evaluation methodology for each waste treatment process is shown in Figure 1. The costs of the four treatment processes of wastes generated during the dismantling of domestic research reactors were re-evaluated and compared with the actual costs. Here, the four treatment processes refer to the hand grinder process of concrete hot cell, lead hot cell, and ventilation facility and the scabling process of underground pipe and ducts.

As a result of the performance verification using the established unit cost data and methodology, the average error was 10%, and when the operating ratio was adjusted in the operating cost calculation part, the average error decreased to about 4.5%. Although the accuracy of the unit cost data and methodology established in this study was confirmed, it is expected that the accuracy will be increased if the data of not only the research reactor but also the commercial nuclear power is analyzed and applied.

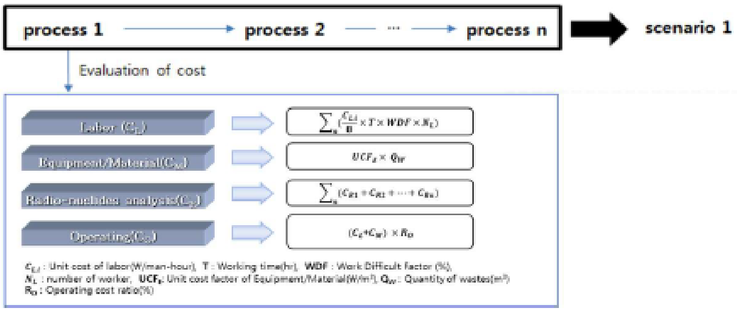

Figure 1. Cost assessment methodology by waste treatment process

Keywords: Cost evaluation, Waste treatment process, Decommissioning

ACKNOWLEDGMENTS

This work was supported by a grant from the National Research Foundation of Korea (NRF), funded by the Korean government, Ministry of Trade, Industry and Energy (No. 2017M2A8A5015143).

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PS5 (T5.6-1033)

A Review on Characterization of Cutting Debris for Worker Safety in the Decommissioning of Nuclear Power Plant

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Cutting and segmentation will be one of the most major activities during the decommissioning of a nuclear power plant. Large components such as Reactor Vessel Internal (RVI), Reactor Vessel (RV), Steam Generator (SG), Reactor Coolant Pump (RCP), other big tanks and vessels that have contained radioactive contaminants will be cut or segmented in order to be dismantled, packaged, and shipped for disposal. In addition, the large amount of pipework from the entire systems requires cutting activities. Therefore, much attention should be taken in factors such as the generation of dust, debris, aerosols, etc. when cutting operations are performed on contaminated parts of components or systems in radioactive areas.

The selection of cutting technologies will generally be related to considerations such as materials of construction, cutting length and width, workability, accessibility, transport and packaging plans, and schedule constraints. Also important is the radiological information about the item or component to be cut, which must already be available before cutting. Debris, which will occur depending on the radioactive contamination level and the cutting plan, will cause exposure to workers performing the activities, especially the internal exposure. Considerations should be also be given to the establishment of measures to treat secondary waste and air contamination generated during cutting works. Typically, thermal and mechanical cutting technologies may be selected for the application in nuclear facilities, and the thermal is more disadvantageous in terms of debris and smoke generation than mechanical one. As shown in Table 1, the cutting technologies used in decommissioning can be summarized in terms of generation of debris, management method, advantages and disadvantages. Also, that provides information on various cutting techniques in order to provide assistance in the planning of cutting operations used during decommissioning.

Table 1. Summary of Generated Debris and Management Method from Major Cutting techniques [1]

Technique	Generated debris	Management method
Thermal – Oxy acetylene gas torches	Fine particles, off-gas, fumes and smoke. Generally particle size: 0.2-0.3㎛.	Fixative for strippable coating on the surface. Use of HEPA filter System with fire protection.
Thermal – Plasma Arc Cutting	Large quantities of aerosol, smoke, dust. Particles size of less than 3㎛.	Fixative for strippable coating on the surface. Use of HEPA filter System with fire protection.
Thermal – Contact Arc Metal Cutting	Similar with Plasma Arc Cutting.	More powerful water filtration equipment is required
Thermal – Laser cutting	To be reduced compared to PAC due to narrower cut kerf widths.	HEPA vacuum and filtration techniques.
Hydraulic - Abrasive Water Jet Cutting	A large volume of water and used abrasive grit.	Underwater filtration and containment of the abrasive materials.
Mechanical - Portable Shearing tool	Minimal debris, no off-gas and fumes.	Use of glove bags or containment tents, localized capture ventilation, including HEPA systems.
Mechanical – Concrete Hammering	Concrete produces large size debris which are readily easy to collect.	Prior to operations fixative for strippable coating. Used to misting device and water.
Mechanical – Concrete Diamond wire Saw	A little debris. A slurry with water cooling.	Possible to cut in dry conditions. Dust can be reduced using a collection system.

***Keywords**: Waste management, Debris, Characterization*

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PS5 (T5.6-1059)

A Study on Public Dose Assessment Approach during Decommissioning of Nuclear Facility for Final Decommissioning Plan (FDP) in KOREA

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Kori unit 1, the Korean oldest commercial NPP, was permanently shut down in 2017 and decommissioning activities will be conducted five years later. KHNP, the licensee, should arrange the final decommissioning plan (FDP) in order to safely carry out the power plant decommissioning. The safety assessment is one of essential part of the FDP because it describes how exposures of the public are kept as low as reasonably achievable (ALARA) below the relevant limits during decommissioning. However, there is no useful assessment methodology for the public dose assessment in decommissioning stage. For this sake, we have carried out the review for the IAEA safety assessment methodology and developed a conceptual diagram that be applicable to the chapter 6.safety assessment part in FDP [1]. Based on this concept, we have analyzed the normal events and abnormal events during the decommissioning, which assessed public dose for each radiation exposure scenario. This paper is to propose considering factors for public dose assessment not only the normal decommissioning activities but also the abnormal events.

Decommissioning stage is different from the operation phase of nuclear facilities in that decommissioning has cutting activities of radioactively contaminated and/or activated components and structures, where radioactive dusts and gas are dispersed into the atmosphere. It is necessary for public dose assessment to estimate the amount of radionuclides released into the atmosphere and the ocean during decommissioning operation. Therefore, it should be assumed in consideration of dismantling activities, residual radioactive inventory, and decommissioning schedule, etc. Fig.1 shows the conceptual diagram to assess the dose to public in the normal situations and abnormal situations. The main consideration to assess the public dose is presented in the Fig.1 as alphabet (a), (b), (c), (d) and will be described details in the poster presentation. Finally, this approach is expected to be applied as the basic methodology for establishing of the FDP.

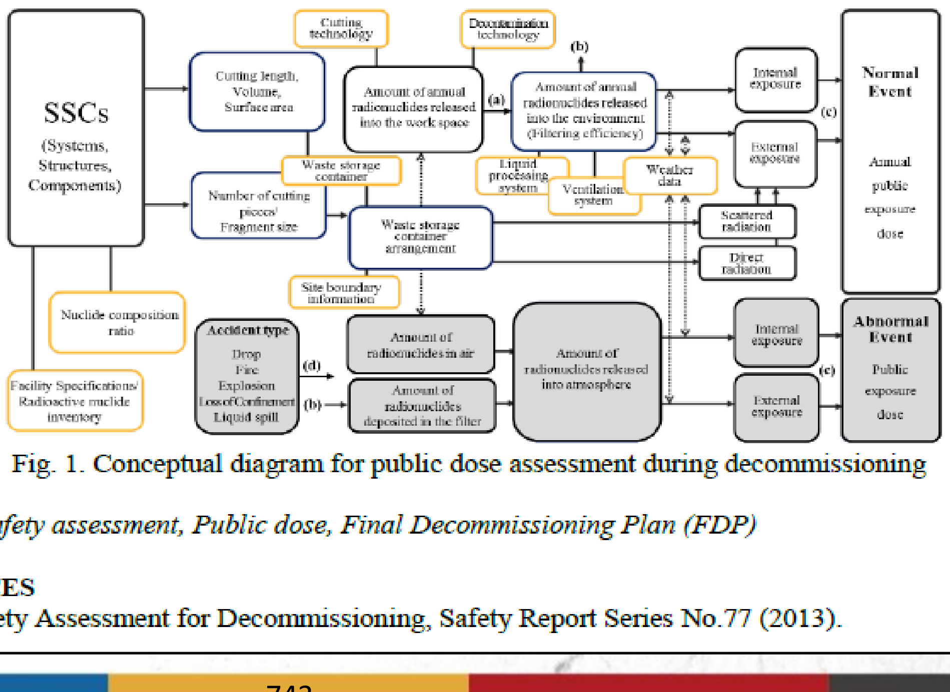

Fig. 1. Conceptual diagram for public dose assessment during decommissioning

Keywords: *Safety assessment, Public dose, Final Decommissioning Plan (FDP)*

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PS5 (T5.6-1061)

Study on IAEA Safety assessment methodology application for Decommissioning of Kori Unit 1

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For nuclear power plants (NPPs) that already have a safety assessment methodology for the operational phase, parts of that safety assessment may be relevant to aspects of the decommissioning safety assessment. However, it is important to recognize that the main target of decommissioning safety assessment is fundamentally different from that of operating the NPPs. As all nuclear fuels are removed or stored in a passively safe form, the existing probabilistic safety assessment (PSA) is not necessarily appropriate method for identifying accident scenarios during decommissioning [1]. Instead, a deterministic approach to safety assessment is the most commonly applied approach methodology to derive accident scenario during safety assessment.

For this sake, we have carried out a review of IAEA safety assessment methodology and adopted a detailed method that be applicable to the decommissioning safety assessment from the according to the Korean Occupational Safety and Health Agency (KOSHA) guideline. Finally, the applicable safety assessment methodologies for deriving potential accident scenarios were reviewed and the detailed approaches are compared with IAEA safety assessment steps giving actual examples in Table 1. It may be helpful to understand the actual methodology of safety assessment at the early phases of planning decommissioning.

Table 1. IAEA safety assessment methodology application by actual examples

I. IAEA Process	II. Applied KOSHA guideline	Examples
1. framework	HAZOP	• Determine a regulatory criteria & requirements (Accident Criteria: 8) • Determine a risk matrix consequence & likelihood criteria (4×4, 5×5)
2. Description of facility / Activities	HAZOP	• Understand the decommissioning activities and facility • Use a facility drawings and
3. Hazard identification / Screening	Checklist	• External hazards such as aircraft, flood, seismic etc. events can be excluded from initial checklist • Radiological, Fire/explosion, Electrical, Chemical, Human etc. hazards are evaluated by experts using the final checklist • Evaluate the potential accident scenarios by checking the checklist for all decommissioning activities. • Derive and list all potential accident scenario
4. Hazard Analysis	HAZOP What if	• Assume the potential accidents from HAZOP with experts • What if fire break out in the radioactive waste storage area? • What if drop the radioactive waste drum during transfer to truck? • What if decontamination waste tank is broken and spilled?
5. Engineering Analysis	HAZOP Risk matrix	• Grouped by the similar derived potential accident list • Determine the consequence & likelihood by accident scenario • Ex) Radiological hazard: cutting the pipe (Consequence: 2, Likelihood: 3, Risk: 2×3 =6) • Ex) Radiological hazard: drop the drum (Consequence: 3, Likelihood: 3, Risk: 3×3 =9) • Select the accident scenarios above 8 score • Assess the "Drop the drum scenario"
6. Identification of measurements	HAZOP	• When cutting the pipe activity, put on the mask, protective clothing. • Take a measurements and lower the score (Consequence: 2, Likelihood: 3→2, Risk: 2×2 =4)
7. Compliance with Criteria	HAZOP	• If the accident scenario score is above 8, then evaluate the scenario.

Keywords: Safety assessment, Public dose, Final Decommissioning Plan (FDP)

PS5 (T5.6-1062)

Study on Kori Unit 1 Non-fuel Waste Treatment Method

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Prior to dismantling Kori Unit 1, all wastes stored in the spent fuel pool (SFP) must be removed. In the SFP, non-fuel wastes such as thimble plugging device (TP), rod cluster control assembly (RCCA), neutron source assembly, in core instrument (ICI) and burnable poison rod assembly (BPRA) etc. are stored in addition to the spent fuel.

The ICI is an instrument for measurement of core neutron flux distribution. It moves through the core or is loaded in the core, if necessary, and provides axial and radial neutron flux distributions within the core. The absorber of RCCA inserted into the fuel assembly (FA) to control the core reactivity or to shut down the reactor consists of Ag-In-Cd. The BPRA inhibits excess reactivity and limits excessive boron concentrations. The B-10 and Gd are used as neutron absorbers.

In Taiwan Tai Power, control rod (CR) was cut and stored in an SFP rack. In the United States, the CR were stored in spent fuel assemblies or in SFP racks. The ICI is stored in sealed vault after cutting or stored in polar crane wall (about 18 months) and then in rad waste area until the radiation level decreases.

In Korea, the decision has not been made yet. However, it will be processed when cutting and dismantling reactor vessel (RV) and reactor vessel internal (RVI). Cleavable components of non-fuel waste will be cut in SFP and packaged with RV cuts or RVI cuts and sent to a radioactive waste disposal facility. However, components containing radioactive materials such as RCCA and BPRA cannot be cut. Therefore, the components that are expected to leak radioactive materials during cutting operations such as BPRA and RCCA will not be cut, but will be transported with spent fuel and stored in an Intermediate spent fuel storage (ISFS).

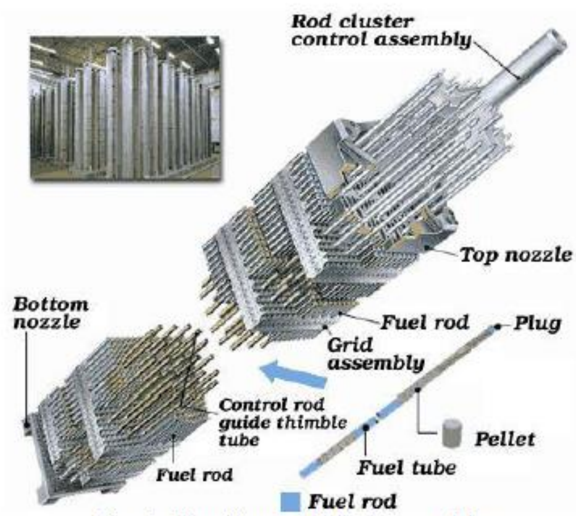

Fig. 1. The figure of fuel assembly

Keywords: Radwaste, non-fuel waste

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PS5 (T5.6-1082)

Radioactivity Monitoring for the Structure of Kori unit 1

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The first nuclear power plant operated in Korea was Kori, which first started commercial operation in 1978, and eventually shut down in June 2015 after 30 years of design life and 10 years of life extension. It was determined by considering comprehensively various factors of nuclear safety, economical efficiency, regional acceptability, power supply & demand, and mid/long term technology. The dismantling method of Kori Unit 1 has been intensively discussed based on overseas cases mainly by many experts in Korea, but there are no concrete plans to date until now. One of the reasons is the lack of radiological data for main equipment and concrete structures in Kori Unit 1. This is an indispensable item for establishing a nuclear dismantling plan, but it is believed to be a lack of policy recognition and technical preparation for nuclear dismantling for the past decades.

In this study, the area of Kori unit 1 was classified and specified to evalutate the radiological characteristics of the structures. Analytical samples were obtained by rubbing with paper on the wall surface (1×1 m), and analyzed radiochemically for activated radionuclides. The contaminated smear samples were heated for about 8 hours at 450 ℃, and add 25 mL of 10 M nitric acid, 25 mL 4 M hydrochloric acid and 50 mL of distilled water. In order to verify the pretreatment recovery rate, 0.3 mL (3 mg Re) of Re was added into the sample pretreatment solution. The ashed samples were dissolved with the acid solution by using the microwave digestion system. Since pretreated solution in the chemical analysis contains various elements and compounds such as Na, K, Li, Al, and Ca, etc, separation procedures of ^{55}Fe, $^{59/63}$Ni, and ^{94}Nb were developed and applied to extract only the radionuclide we want. A recovery evaluation of Fe, Nb and Ni for all of the samples was carried out using a prepared simulated solution whenever the radionuclides (^{59}Fe, $^{59/63}$Ni and ^{94}Nb) from the radioactive samples were individually separated. As a result of evaluating the separation reliability, it was found that all standard deviations were less than 3%.

Figure 1. Sampling and radiochemical analysis

Keywords: Characterization, Radioactivity distribution, decommissioning, radionuclide separation

ACKNOWLEDGMENTS

This work was supported by the Ministry of Science and ICT (MSIT) of South Korea.

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PS5 (T5.6-1088)

A Review on Calculation Methodology on Gaseous Discharge During Decommissioning of Nuclear Power Plants

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Kori unit 1, the Korean oldest commercial NPP, was permanently shut down in 2017 and decommissioning activities are being conducted. KHNP should summit FDP (Final Decommissioning Plan) in accordance with regulations of Nuclear Facilities Decommissioning Plan, such as Nuclear Safety and Security Commission Notice 2018-10 "Which is the guideline for the FDP". As the Notice, the FDP consists of eleven chapters. Chapter 9 refer to the radioactive waste management which includes solid, liquid, gaseous, mixed waste and operational radioactive waste. In this paper, we have analyzed the method of calculating of the gaseous discharge and used as input basic into the Kori-1 FDP.

The amount of gaseous wastes depends on cutting technology, number of cutting pieces, cutting length, etc. Especially, each cutting methods such as plasma arc, thermal cutting, mechanical cutting has respectively different Kerf width and dispersion ratio during cutting activities. And the Kerf length depends on the type of the container in which the component will be stored. Shapes of components such as piping and ducts also affect the Kerf lengths. Therefore these factors need to be considered for the evaluating the amount of gaseous wastes. To calculate the quantity of generating aerosol, the following information is required and total amount of generating wastes (G_{ij}) is simply expressed in the below equation (1) [1].

$$G_{ij}(Bq) = \sum(C_i \times K_j \times D_{ij}) \cdots\cdots\cdots\cdots\cdots (1)$$

Where,

- G_{ij} of radionuclide i and cutting tool j
- Concentration of radionuclide : Ci (Bq/cm^3 or Bq/cm^2)
- Kerf volume / surface : K_j (cm^3 or cm^2)
- Dispersion ratio of each cutting tool : D_{ij}

The source term data were obtained from the Characterization Survey for Kori unit 1. The report categorized major components (RV, RVI, SG, PZR, and RCP) and others (Pipe, Valve, Pump, Scabbling Concrete, etc.) for decommissioning systems and structures. In case of Kori-1, the total number of components or devices is approximately 16,000 [2]. The components are classified into 6 shapes according to the type (cuboid, circular piping, large object with tube, cylinder, plate, hollow prismatic column).

In addition, definitions and calculation methodologies for Kerf volume, dispersion ratio and cutting length are needed. The methodologies and conclusions (As the detailed factors are decided, the accuracy of the result) not mentioned in the abstract will be presented in a poster presentation.

This gaseous discharge calculated through this methodology will be used as basic data for radiological environmental impact assessment and public dose during decommissioning of NPP such as Kori-1 and Wolsung-1.

Keywords: *Gaseous Discharge, Decommissioning, Final Decommissioning Plan*

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PS5 (T5.6-1104)

A Review on Considerations of the Segmentation Process for the Decommissioning Heavy Water Reactor Type

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Decommissioning is the final phase in the life cycle of nuclear facilities after operation. Decommissioning of nuclear facilities is of great interest because of a large number of facilities which were built many years ago. Kori Unit 1 was permanently suspended in June 2017 for the first time in Korea and Permanent shutdown for Wolsong Unit 1 was finally permitted in December 2019 by Korea Nuclear Safety and Security Commission (NSSC). Wolsong Unit 1, the first decommissioning heavy water reactor that has a policy of the direct decommission has much attention how to dismantle and control the radioactive waste. KHNP (Korea Hydro Nuclear Power) will prepare for appropriate dismantling technologies in a few years. Unlike PWR (Pressurized Water Reactor), the Pressurized Heavy Water Reactor (PHWR) is specifically composed of many components such as caldaria tubes, pressure tubes, calandria vault structures, reactivity mechanisms etc. Therefore, KHNP needs to establish a specific calandria segmentation strategy and develop thorough analysis of multiple conceptual strategies based on the PHWR characteristics. Accordingly, in this paper, the segmentation methods were proposed as part of the conceptual segmentation strategy and technologies, selected tools as table 1. The recommended strategy is based on results of characterization assessment, minimizing technological risk, utilizing commercially available equipment, and the development of remote tooling where high radiation precludes manual operations. In Korea, this study will be based on insitu application to make a practical segmentation process of heavy water reactor type.

Table 1. Conceptual Segmentation Process to consider the Decommissioning Heavy Water Reactor Type

Segmentation step	Process considered	Technology considered
Characterization assessment	- Select radiation levels for location of the calandria from Characterization assessment - Make a segmentation plan(mapping to cut)	- Select radiation levels to make segmentations by cutting
Preparation	- Install HVAC & temporary devices to collect cutting debris	- Select fuel channel removal tools - Reactivity mechanism & Reactivity deck removal tools - Perform an actual training to cutting using Mock-up for high radiation working area
Segmentation Step	- Review appropriate tools according to component (caldaria tube, pressure tube, calandria vault etc.)	- Diamond wire saw, Oxygen torch, Rotating saw, Torch processes, Abrasive water jet, Contact Arc Metal Cutting (CAMC), Laser
Waste management	- Consider an appropriate container according to radiation level of cutting segmentations - Select disposal methods of Radioactive waste	- Select an appropriate container within a criteria of radiation level

Keywords; Segmentation strategy, dismantling technologies, Segmentation Step

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PS5 (T5.7-0291)

Proposal for indicators to compare the harmfulness of radioactive materials and waste

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French National Plan for Radioactive Materials and Waste Management (PNGMDR) asked to the French Institut for radiation protection and nuclear safety (IRSN) to develop a method and indicators in order to compare the harmfulness of waste and materials. The simplified methodology proposed is based on three exposure situation and concerns population and ecosystems. The part of methodology discussed here concerns human exposure pathways.

The three exposures situations are defined to represent generic situations in order to cover a maximum of real exposure situations. The choice of these exposure situations is derived from international publications as for example those describing the method used for the derivation of Activity Concentration Values for Exclusion, Exemption and Clearance [1] or D-Values [2].

In the first situation a person is supposed to carry out an occupational activity in an enclosed place wher stands a undamaged package of radioactive waste or materials. The exposure pathways associated with the regular occupation of the enclosed place are then external exposure due to the radiation emitted and inhalation of radon due to the potential exhalation from the waste or materials;

In the second situation, the waste or materials content is considered to have been suddenly dispersed into the enclosed atmosphere resulting in an accidental exposure. Exposure pathway is then restricted to inhalation of suspended dust;

In the third situation, radiological and chemical substances contained in waste or materials are supposed dispersed into the environment: A group of population leaves and eats foodstuffs coming from this environment. Exposures pathways are ingestion of contaminated foodstuffs (vegetables, fishes and so on) and drinking water.

The aim of this presentation is to describe these scenarios and the associated hypotheses as well as early results.

Keywords: *harmfulness, radioactive waste, radioactive materials*

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PS5 (T5.7-0324)

Shielding Study for Chinese First High Activity Disused Sealed Sources Conditioning Mobile Hot Cell

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Abstract: Spent or disused high activity radioactive sources should be shielded in the whole process of conditioning, because of high dose rate and risk. The shield experiment devices for high activity sources conditioning study was designed and built up. Different material and consideration was studied for shielding the spent or disused high activity radioactive sources. After our study, suitable method was carried out. It could be an good example for radioactive waste safety .

(Key Words: high activity， spent radioactive source， conditioning， shielding

PS5 (T5.7-0357)

An innovative platform allowing digitization of operative Radioprotection measurements and to characterize NORM, TENORM and nuclear waste

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Abstract: The process of dismantling and decommissioning nuclear infrastructure and the periodic control of industrial infrastructures accumulating NORM increasingly demands methods for a full traceability of devices and measurements to improve quality management and operational safety.

The utilization of measurement systems and equipment from disparate sources, often unable to share information with one central database, have forced Operators to devise ad hoc management systems. These management systems are often based on complex and sometimes even incomplete or inaccurate logbook notations. The result is a complex procedure burdened by poor QA/QC and an increased likelihood of errors.

CAEN Proposes a platform which relies on an innovative handheld instrument combining state-of-the-art radiation measurement capabilities with read/write UHF RFID tagging, integrating a color camera, audio recorder, GPS and UWB localization for both outdoor and indoor positioning. Taking advantage of this system D&D or radioprotection operators can easily characterize radioactive sources and start recording the digital information at the earliest stages of their activities. All the information can be then uploaded, safely stored and processed by a customizable database software framework.

All the digital information can be the retrieved in a second phase when contaminated tools or scrap material shall be collected and properly selected for the subsequent disposal. On-field operators can then be equipped with dedicated, ruggedized and portable RFID readers.

PS5 (T5.7-0364)

Recovery of Uranium from Residue generated during Mo-99 production, Using Organic Solvent Extraction

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During the dissolution process of the irradiated uranium target plates used for production of the medical isotope Mo-99, the uranium and some fission products are precipitated as a residue of mixed hydrated oxides. The aim of this research was to evaluate organic extraction ligands that can operate in alkaline media to remove uranium from the nuclear waste and the objective was to characterize the most effective organic solvent for extracting uranium only, from alkaline media. Uranium oxide was dissolved in ammonium carbonate solution to form the uranium tri-carbonate aqueous feed solution. Different concentrations (0.2 M, 0.5 M, 1 M and 1.5 M) of ammonium carbonate and different concentration (0.005 M, 0.025 M, and 0.01M) of uranium in volume ratio (5:1, 2:1, 1:1, 1:2, and 1:5) with 15% Aliquat 336 in Toluene as organic solution were evaluated. It was found that 0.2 M of ammonium carbonate and 0.01 M uranium at volume ratio of 1:5 to be the highest extraction parameters that can be used to extract uranium with a percentage of 98%.

Keywords: *alkaline media, Aliquat 336, organic solvent extraction*

PS5 (T5.7-0487)

The 15th International Congress of the International Radiation Protection Association Special Techniques to improve accuracy of radiometric characterisation based on ISOCS

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ISOCS is an established tool used for gamma spectrometry. It allows physical representation of complex geometries and mathematical calculation of the calibration function while avoiding the need for radioisotope standards. Additional special features have been developed based on ISOCS to improve measurement accuracy.

ISOCS provides a large range of templates to generate the geometry for most nuclear waste components in nuclear facilities. ISOCS is user friendly as to the generation of models based on a large range of templates. Actually most of Mirion's Non Destructive Assay (NDA) systems calibration functions are mostly based on ISOCS and are validated by real sources. Measurement accuracy satisfies many of the measurement requirements among worldwide sites' regulations.
In addition, extended service tool is able to generate very complex geometries in ISOCS for specific geometry measurements. These functions refine the modelled geometry compared to the real objects and decrease the measurement systematic errors.

Moreover, ISOCS Uncertainty Estimator (IUE) is a tool of ISOCS which allows estimating measurement uncertainty. Specific functions Advanced IUE (AIUE) have been developed to minimize measurement uncertainty. These functions optimize ISOCS models using Figure Of Merit (FOV) Minimization by means of two types of algorithms to the real data. These functions help drop dramatically measurement uncertainty and thus reduce the cost of waste storage.

In this paper, we demonstrate through ISOCS modelling of real object data test, the agreement of the efficiency calculation by comparing ISOCS to MCNP calculations. Compare to the previous demonstration, we will show the new result of improvement of measurement accuracy to real data using ISOCS and specific function AIUE. In addition, this paper will show the application for real waste assay projects.

***Keywords**: ISOCS, improvement of measurement accuracy, AIUE*

PS5 (T5.7-0780)

Management of Disused Radioactive Sources at Sint-Elisabeth Hospital, Willemstad, Curacao

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Sint-Elisabeth Hospital (SEHOS) was for many years the main public hospital in Curacao, located in the Otrobanda district of Willemstad. A new hospital, the Curacao Medical Center was built and started to operate at the end of 2019. Medical services are now provided in the new hospital. Facilities of the old hospital (SEHOS) will be used for other purposes or demolished. Unknown amounts of legacy radioactive sources used for brachytherapy were stored at SEHOS, including thousands of Ir-192 seed sources, around 25 Cs-137 tubes and an unknown number of very old Ra-226 brachytherapy sources.

Curacao requested assistance from the IAEA to improve the situation regarding the disused radioactive sources (DSRS) at SEHOS, before the Radiotherapy Department was completely moved to the new Hospital. An IAEA Expert mission was conducted in October 2019 to provide support for the recovering, characterization, conditioning and safe storage of the disused brachytherapy sources.

The DSRS were initially kept in two areas: (1) at the Radiotherapy Department and (2) at an underground store. The 93 containers with the Ir-192 seed sources initially stored in 22 packages at the brachytherapy room and 53 packages at the underground store were checked and consolidated in 23 packages. Records were completed. The new packages were transferred to the teletherapy room, which was identified as the most suitable area to be used for temporary storage of DSRS at SEHOS. Radiological controls were carried out in the former storage area of the Ir-192 sources, as well as in the empty packages. Forty seven (47) Ir-192 sources were recovered and safely stored. No radioactive contamination was detected.

Seven (7) Cs-137 disused radioactive sources were recovered from the storage containers and devices (applicators), characterized and conditioned in a stainless steel capsule. Additional Cs-137 brachytherapy sources were not conditioned because they were still in used. Conditioned capsule was placed in a proper shielded container and transferred to the teletherapy room for safe storage. The record of conditioned DSRS was prepared, with detailed information on each radioactive source and the devices from where they were recovered.

The Ra-226 DSRS were not recovered and handled yet, as loose radioactive contamination was detected in the storage area. Radium-226 sources remained in a container placed in a hole under a concrete bunker located in the underground store. A Decommissioning Plan of the underground room should be prepared before starting the operations. The plan includes the recovery, characterization and conditioning of the Ra-226 DSRS; the characterization of radioactive contamination in all areas; the dismantling and decontamination activities; the management of generated wastes; the emergency plan and the final radiological survey.

ACKNOWLEDGMENTS

The authors want particularly to thank Mr. Lucien Zuiverloon and other personnel from SEHOS for their support for the implementation of the DSRS management operations.

PS5 (T5.7-0807)

^{3}H and ^{14}C Radioactivity Analysis of Paper Box from Nuclear Fuel Package

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Though the radioactivity of the nuclear fuel itself is sufficiently low, low neutron flux exists due to spontaneous fission of the ^{235}U and ^{238}U. In the aspects of criticality safety, it is not considered because the nuclear flux is sufficiently low and there is now moderator around the fresh fuel to support the chain reaction. Nevertheless, it is hard to quantify how much neutrons occur, neutron activation of nitrogen and production of ^{3}H and ^{14}C is possible [1]. The ^{3}H and ^{14}C radioactivity analysis of paper box from the fresh nuclear fuel package was carried out in this study. 22 different paper box samples were analyzed which were categorized by three: "BD", "BOX" and "EC". Oxidation with combustion method was adopted for pretreatment of the paper box sample because large amount of sample should be processed. The high temperature furnace (Pyrolyser-6-trio, RADDEC) was used and preset temperature control program was used where the temperature was increased up to 900°C for complete oxidation of ^{14}C. 2.5±0.2 g of each sample was loaded on the sample boat. The recovery rates of ^{3}H and ^{14}C of the furnace were estimated by 90.3±3.3 and 87.9±3.0%, respectively. ^{14}C is detected only from four samples (from the BOX 20, 22, 23 and 25). But the concentration of ^{14}C is very low and not concerning. However, ^{3}H is detected in all samples, where the lowest ^{3}H concentration was 7.8±0.6 Bq/g. Even the maximum concentration was 283.5±20.9 Bq/g which exceeds the limit of clearance level of ^{3}H (100 Bq/g).

Table 1. Analyzed ^{3}H and ^{14}C concentrations of the paper box samples

Sample ID	^{3}H Concentration (Bq/g)	^{14}C Concentration (Bq/g)	Sample ID	^{3}H Concentration (Bq/g)	^{14}C Concentration (Bq/g)
BD1	22.4±1.7	<0.070	BOX22	177.7±13.1	0.099±0.009
BD2	22.0±1.6	<0.057	BOX23	283.5±20.9	0.101±0.010
BD3	10.7±0.8	<0.070	BOX24	128.2±9.4	<0.051
BD4	7.8±0.6	<0.066	BOX25	191.2±14.1	0.112±0.010
BOX19	93.9±6.9	<0.056	EC23	27.0±2.0	<0.071
BOX20	148.1±10.9	0.078±0.007	EC24	196.6±14.4	<0.048
BOX21	107.6±7.9	<0.061	EC25	17.0±1.3	<0.046

Keywords: *^{3}H,* ^{14}C, paper box, nuclear fuel package

ACKNOWLEDGMENTS
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PS5 (T5.7-0822)

A Review of Transfer Parameters of I, Cs and Pu (2) -Concentration Ratios in Freshwater and Marine Fish-

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In the environmental radiation dose assessment for nuclear waste management, water-to-fish concentration ratio (CR) is of our great concern because Japan is one of the biggest fish consuming countries in the world. In many countries, transfer parameter values for dose assessments are chosen from the international data sources such as the IAEA Technical Reports Series (TRS) and/or scientific literatures reporting on-site data. In Japan, data in the IAEA TRS and those obtained outside Japan are mainly used; these data are originally observed in many years ago, e.g. in 1960s and 1970s. There are more data available now, therefore, it is important to compare between old and new data sets to clarify the reliability of the data. Thus, in this study, we carried out literature survey (1) to check how these data were obtained and why they were selected, and (2) to compare international data and recently available data. We used the following three search engines, Google scholar, Science Direct, and J-stage, with keywords "iodine (I), caesium (Cs) or plutonium (Pu)", "concentration factor", "radioactivity", "fish" and "freshwater, river, lake, marine, or sea" for this survey. We did not use CR data for Cs observed in Japan after the accident in the Fukushima Daiichi nuclear power plant (FDNPP) because the water-to-biota systems were not in equilibrium-conditions in many cases.

For Cs, Figure 1 shows the results of the literature values for the fish CR in freshwater (FW) and brackish water (BW), and the average, minimum, and maximum values of TRS-364, 422, 479 and Safety Reports Series (SRS) -19. The results showed that the CR values of whole body (CR_{whole}) and edible part (CR_{edible}) of fish in FW were within the ranges of TRS series. CR_{whole} values in the BW from the literature were 1-2 orders of magnitude higher than those in TRS series however the reason was not clear. Unfortunately, there is no CR-BW data in Japan, therefore it is desirable to obtain CR-BW to provide reliable data for environmental dose assessments.

For CR_{edible} in seawater (SW) before FDNPP, we found four publications. The CR_{whole} values in SW tend to be lower than those in FW and BW. We compared CR data for I and Pu in the same manner as Cs.

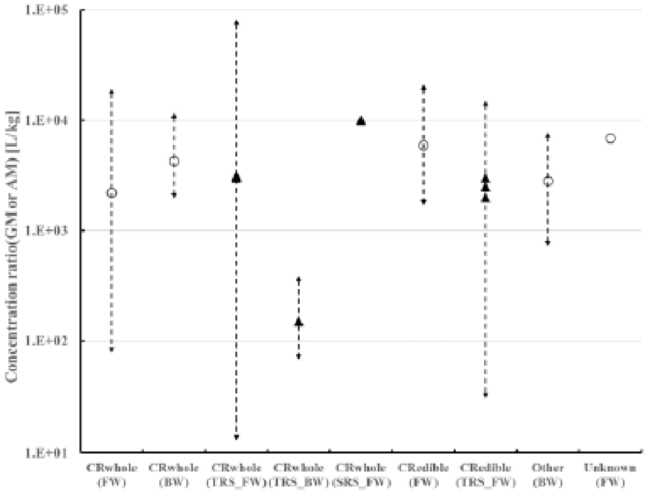

Fig.1 CR values of Cs in FW and BW. Target body parts are whole body (whole) and edible parts (edible). ○ = Literature value, ▲ = IAEA TRS[1] and SRS-19[2].

Keywords: Water-to-fish concentration ratio, Environmental dose assessment, Parameter values

ACKNOWLEDGMENTS

The authors thank to Dr. Tomoyuki Takahashi (Kyoto University) for his valuable comments.

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PS5 (T5.7-0825)

A Review of Transfer Parameters of I, Cs and Pu (3) - Concentration Ratios of I and Pu in Marine Biota-

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Number of data for seawater to marine biota concentration ratios (CR) of iodine (I) and plutonium (Pu) are limited in Japan.[1] For crustaceans, and molluscs (cephalopods and shellfish), the numbers of CR data were usually low for both elements(N=1-5); for fish muscles, relatively large numbers of CR data were found in previous literature,[1] i.e., 11 for I and 15 for Pu. Because fish habitats may cause different CR values, it is better to classify fish species into fish types, i.e. benthic and pelagic, but numbers of data were too small. To provide reliable recommended CR values for the long-term dose assessment, it is necessary to have more data. In this study, we carried out a data survey and calculated CR values using global fallout Pu and stable I. Activity concentration data of $^{239+240}$Pu in seawater and biota were selected from open data sources (the Environmental Radiation Database, http://search.kankyo-hoshano.go.jp/servlet/search.top). The CR-Pu was obtained by dividing Pu in biota (Bq/kg wet) by that in seawater (Bq/L) collected within 10 days before or after the biota sampling date. If several Pu concentration data in seawater samples were available for the corresponding period, then the average concentration was used for CR calculation. For the case of I, stable I concentration data compiled in the Food Composition Database, (https://fooddb.mext.go.jp/) were used. We applied I concentration data in Japanese coastal seawater.[2]

Number of calculated CR data of Pu in algae was the largest (N=95); the calculated data distributed log-normally. We estimated that the other groups also would show log-normal distributions; thus, geometric mean (GM) values were calculated and these data were compared with the recommended values by IAEA[3] (Table 1). Except algae, GM of CR were similar to those recommended by IAEA. The CR values in algae for I and Pu by IAEA were obtained from brown algae; CR values would change among different algae groups, thus, classification of algae type would be necessary for analysis. This study could provide more CR values than before, but data are still limited for some biota groups, i.e. crustacean. Further data collation is necessary for these specific biota groups.

Table 1. Concentration ratios of Pu and I from seawater to biota

Group	I (This study)	I (IAEA[3])	Pu (This study)	Pu (IAEA[3])
Algae	150 (N=32)	10000	970 (N=95)	4000
Crustacean	8.1 (N=4)	3	250 (N=3)	200
Cephalopods	10 (N=7)	-	110 (N=9)	50
Molluscs (except cephalopods)	21 (N=7)	10	1300 (N=34)	3000
Fish (muscles)	4.3 (N=34)	9	20 (N=10)	100

Keywords: *Water-to-biota concentration ratio, Long-term dose assessment*

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PS5 (T5.7-0831)

A Review of Transfer Parameters of I, Cs and Pu (1) - Feed Transfer Coefficients in Cow's Milk-

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Internal dose from the ingestion of contaminated agricultural products is an important portion in the radiation dose assessment for nuclear waste management. It is reported that cow's milk is the biggest contributor of internal dose in the early stages of the Chernobyl nuclear accident[1]. Cow's milk is consumed by infants; they are relatively sensitive to radiation compared to adults[2]. The radionuclide transfer coefficient from feed to milk (C_{ft}_milk) is therefore also one of the important environmental transfer parameters for radiation dose assessment for discharged radionuclides. Many countries use C_{ft}_milk values compiled in the IAEA Technical Report Series in mathematical models for dose assessment. These data were originally from old data sources; however, to keep the transparency of the dose assessment results, it is necessary to clarify the accuracy of the data, and it is also important to compare them with recently available data. In this study, therefore, we carried out literature survey by tracking sources listed in the IAEA reports, and compared those data with recently published data obtained by research institutes in many countries and regions.

To track the data sources, we used review reports on parameter values for radiation dose assessment of waste management published in Japan and other countries. For new data survey, we used Google scholar and J-stage (the largest scientific publication search engine in Japan) to collect papers and laboratory reports. The selected key words for literature survey were, "Transfer coefficient or Concentration ratio", "Milk", and "Animal or Cow". We focused on iodine (I), caesium (Cs) and plutonium (Pu) because their long-lived isotopes.

Most of the parameter values in IAEA reports were based on laboratory studies or estimated using metabolic models of nuclides. In Japan, we found several reports on C_{ft}_milk of radiocaesium observed after the Fukushima Daiichi nuclear power plant accident, but the data of I and Pu were scarce. From recent publication on C_{ft}_milk observed other countries, we found some data for iodine; however, we could not find data for Pu. The latest parameter values in IAEA TRS-472[3], compiled C_{ft}_milk data for I (N=104) and Cs (N=288); Pu data were from review and model studies[4]. As the results of the data survey, the parameter values of I and Cs were similar among the data sources; however, Pu values differed by four orders of magnitude among sources (Fig. 1). Among three elements, C_{ft}_milk values for Pu were 2-6 orders of magnitude lower than those for I and Cs. The consumption amount of farm animal products in Japan is smaller than those in European countries, therefore, several times differences in parameter values has little effect on the dose assessment results; however, the large variation of C_{ft}_milk for Pu could cause large uncertainty of the assessment results.

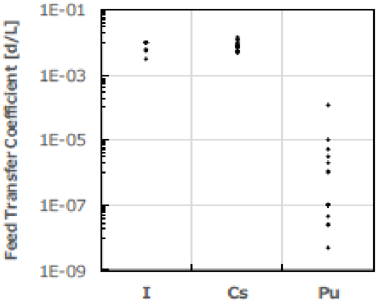

Fig. 1. Comparisons of reported feed to milk transfer coefficients of I, Cs

Keywords: Transfer Coefficient, Feed-to-Milk, Environmental dose assessment

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PS5 (T5.7-0918)

A Study on the Reducing Radiation Exposure Dose to Worker and Safety of Disposal of d-cRWW Granules in NPP

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Domestic nuclear power plants(NPP) treat the concentrated radwaste water(cRWW) by drying with Concentrate Treatment System(CTS) for polymer-adding solidification. At this way, not only approximately 30% of dried cRWW(d-cRWW) granules do not meet the domestic disposal particle size standard (1 ~ 13mm), but also d-cRWW granules are scattered, which may cause internal and external radiation exposure due to worker's breathing. Therefore, in this study, mimetic samples similar to d-cRWW granules were manufactured and then pelletized by compression molding to reduce radiation exposure of workers and to ensure disposal safety. At the first trial of pelletizing, the strength of tablets was weak and capping, laminating and sticking occurred when using the mimetic sample only. Even if the strength and shape of the specimen are good when pelletizing mimetic samples with a lubricant, for reducing the volume of radwastes and ease handling, the upper and lower punches were coated with diamond, dies were diamond coated and tapping, so the shape of tablets was good, and its strength was measured in the range of 28.8 to 35.7 kgf higher than the paraffin solidified strength standard. As such, it is deemed possible to secure NPP disposal safety if d-cRWW granules is manufactured in pellet form by compression molding, and if performed by automated process, it is expected to contribute to the reduction of radiation exposure dose to workers.

Table 1.Compression Test Results of Pellet

Specimens	Appled Pressure(kgf)	Thickness of Pellet(mm)	Compression Strength(kgf)
1	250 ~ 300	6.0	28.8
2	500 ~ 550	5.7	30.1
3	500 ~ 600	5.5	35.7
4	500 ~ 650	6.0	33.6
5	900 ~ 1000	5.8	31.2

Fig. 1. Pellets Used for Strength Measurement

Keywords: d-cRWW, Reduction of Exposure Dose, Safety of Disposal

ACKNOWLEDGMENTS
This study was conducted as a research and development project for the Small Business Cooperation of the KHNP.

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PS5 (T5.7-0968)

Disposal of the sealed radioactive sources generated in Cuba using isolation in deep well (borehole disposal concept)

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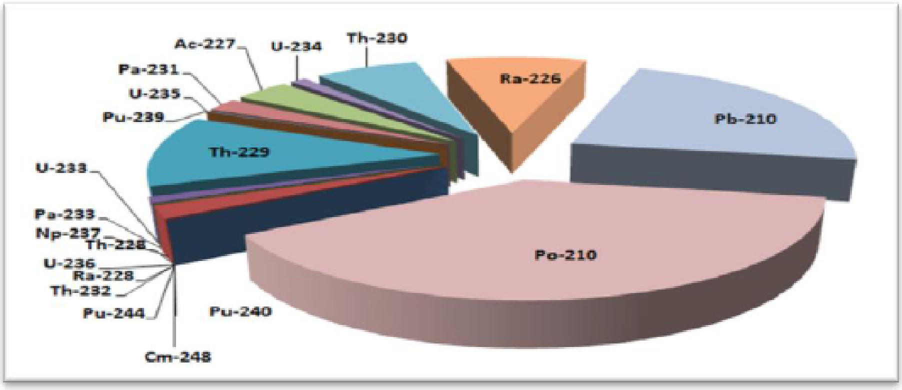
The safety isolation of the sealed radioactive sources is a national priority, assisting to the high volumes stored temporarily and to the continuous generation of these types of radioactive waste. The unfavorable geologic conditions of Cuba complicate the site selection to prevail the extremely permeable and unstable carbonated rocks. A variant of sure disposal of these sources is shown applying the concept of deep well summoned in a sedimentary geologic formation and is carried out the safety assessment of this proposal. For the evaluation the "software "Borehole Disposal Concept" Scoping Tool v2.0" is applied (BDC), as calculation tool. The results show particularities of the Cuban design for the disposition in well (BOREHOLE), makes a call of attention and offers recommendations for the site selection and the best design of disposal of hazardous waste in Cuba. The obtained results also show the impact of the final dose, according to the inventoried radionuclides that arrives to the well of water consumption, assumed by the tool, with location to 100 m of the borehole, to see figure 1.

Figures 1. Total dose of inventories arrived to the water consumption well.



Keywords: borehole, site selection, radioactive sources

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PS5 (T5.7-1014)

Development of Shielding and Containment Requirements for Design Approval of Spent Fuel Storage Cask

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The Public Engagement Commission on SNF management (PECOS) published the recommendation report on the spent fuel management of Korea in June, 2015. Based on this report, the national basic policy for high level radioactive waste management of Korea was established and promulgated, taking into consideration the national/international trends on policy and technology development in 2016. However, Korea government decided that the existing HLW management policy reconsideration through public engagement in 2017. Therefore, nobody knows when the siting of spent fuel storage facility can be decided in Korea but the demand of industrial side on the spent fuel storage cask design approval has been increased. In response to this demand, it is under way to introduce a design approval of spent fuel storage cask into the nuclear safety act.

The aim of this study, as part of the development for regulatory requirements of design, material, structure and performance of storage cask, it is that the development of the shielding and containment requirements of storage cask.

During normal operations and anticipated occurrences, dose calculations based on the allowable leakage rate must demonstrate that the annual dose equivalent to any real individual who is located at the exclusion area boundary does not exceed the criteria of Article 16 (2) 1 and 2 of the standards on radiation protection, etc (the notice of Nuclear Safety and Security Commission No.2019-10)

For any design-basis accident, dose calculations based on the allowable leakage rate must demonstrate that an individual located at the exclusion area boundary does not receive over the 50 mSv of effective dose, 150mSv of Equivalent dose to eye, and 500mSv of equivalent dose to hands, feet, and skin, respectively.

These requirements are not based on the surface of storage cask but based on the exclusion area boundary of the spent fuel storage facility where storage cask will be installed.

Several storage casks will be installed in the onsite or offsite (centralized interim storage) of nuclear power plant and there are no limit on the number of the installed storage cask

This means that designers should design the shielding and containment of storage cask through the most conservative assumption scenario (number of storage cask to be installed, site characteristics, and so on).

***Keywords**: spent fuel storage cask, design approval, Shielding and Containment Requirements*

PS5 (T5.7-1029)

Sampling of depleted UF6 : KAERI case on the safety aspects

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Depleted UF_6(D-UF_6) is byproduct of enrichment process. Although Republic of Korea(ROK) does not have any enrichment facility, KAERI has D-UF_6 which derived from enrichment activities performed by the US for ROK [1]. UF_6 are stored in the 48Y cylinders and a 30 inches cylinder.

D-UF_6, itself is alpha and weak gamma emitter with low risk. However, if leaked, they react with water vapor in the air to form hydrogen fluoride (HF) and uranyl fluoride (UO_2F_2) which are chemically toxic. D-UF_6 is relatively high reactive material that needs to be transformed into a stable form of oxide materials such as U_3O_8 or UO_2 for long-term storage.

$$UF_6 + 2H_2O \rightarrow UO_2F_2 + 4HF$$

The D-UF_6 stored in the KAERI Neutron Science Auxiliary Building will also be converted into a stable oxide form. For this purpose, a sampling system was designed to check the state of the material inside the cylinders. In the sampling design, the scenarios of possible accidents for safety evaluation were established, and the countermeasures and mitigations were identified. In this research, six risks were identified as follows. This result will be used to the license for de-conversion of D-UF_6 in KAERI.

Table 1. Risk of Sampling Systems [2].

Items	Risks
1. Risk of dispersion of radioactive and toxic substances	Leak in box 1 when a cylinder is connected or disconnected/ loss of leak tightness in a UF_6 pipe seal
2. External Exposure Risk	Human errors
3. Fire	Fire risk analysis
4. Handling risk	Failure of a handling apparatus/human errors
5. Chemical risk	Risks of liquid nitrogen/Risks of anoxia
6. Risk linked to loss of utilities	Loss of ventilation/Loss of the gaseous effluents treatment system

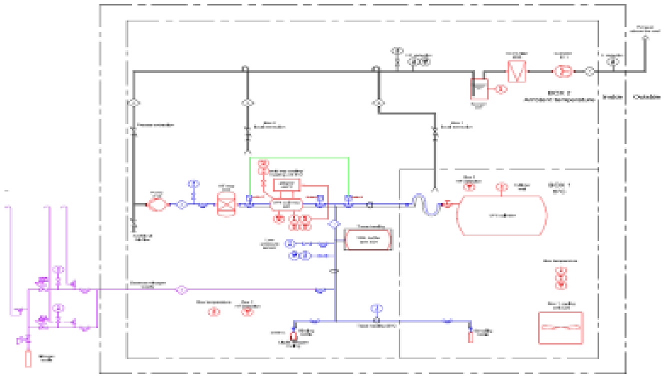

Fig 1. Schematic Diagram of D-UF_6 Sampling System [3].

Keywords: Depleted UF_6, UF_6 sampling, Safety evaluation

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PS5 (T5.7-1041)

Reduced order modeling of near-field THMC processes for nuclear waste repository performance assessment

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Performance assessment (PA) of geologic radioactive waste repositories requires three-dimensional simulation of highly nonlinear, thermo-hydro-mechanical-chemical (THMC), multiphase flow and transport processes on large scales (many kilometers) and over long timespans (tens to hundreds of thousands of years). This work focuses on integrating the effects of a near-field geomechanical process, buffer swelling, between fully-coupled THMC simulations (using TOUGH-FLAC) and THC simulations (using PFLOTRAN) to reduce dimensionality and improve computational efficiency.

The near-field THMC process caused by thermal and hydrological behaviors of pore fluids and subsequent geomechanical perturbation can impact the disturbed rock zone (DRZ) surrounding the drifts of a shale-hosted deep geologic repository that links heat/fluid flow and reactive transport between the engineered barrier system (EBS) and the host rock. Simulations presented here use PFLOTRAN to model a single waste package in a shale host rock repository, where re-saturation of a bentonite buffer causes the buffer to swell and exerts stress on a highly fractured disturbed rock zone (DRZ). Compressing these fractures results in reduced permeability, which could have implications for radionuclide transport and exchange with corrosive species in host rock groundwater that could accelerate waste package degradation.

Keywords: *Repository System Analysis, Near-field process, THMC coupling*

ACKNOWLEDGMENTS

This work was supported by the US Department of Energy (DOE) Office of Nuclear Energy, through the Office of Spent Fuel and Waste Science and Technology (SFWST), within the Office of Spent Fuel and Waste Disposal (DOE NE-8). SNL is managed and operated by NTESS under DOE NNSA contract DE-NA0003525.

PS5 (T5.7-1051)

A Study on Characteristics of Spent Resin from Domestic PWR

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During the operation of nuclear power plant (NPP), various radioactive wastes are generated. They are classified as dry active waste, boron concentrates, spent resin, spent filter, etc. The resins are primarily used to purify the coolants in the loop. Roughly, the ratio of spent resin with respect to total radioactive waste is around 14%. The spent resins were solidified using a cement solidification agent in early stage of NPP operation in Korea. Recently, the spent resins are sent to the high integrity container (HIC) and dried using spent resin drying system (SRDS). In the united states, the low activity spent resins, usually class A, are incinerated. The high activity spent resins, usually class B and C, are dried and inserted in HIC. In Germany, the spent resins are dried, compacted, and packed in metal containers.

The expected volume of spent resins from two unit is shown in table 1. The spent resins are generated from chemical volume control system (CVCS), SFP/SG demineralizers, and liquid waste management systems. The expected total volume from two units is 1,800 ft^3 per year. The important nuclides in spent resins are Co-58, Co-60, Cs-137, and Sr-90. In this study, the expected specific activity, waste classification of the spent resins, and specific activity trend with respect to time will be discussed.

Table 1. Expected volume of spent resin from two units

Spent Resins	Number	Volume(ft^3)	Total Volume(ft^3)
Purification Ion Exchanger	4	36	144
Deborating Ion Exchanger	2	36	72
Preholdup Ion Exchanger	2	36	72
Boric Acid Condensate Ion Exchanger	2	34	68
Spent Fuel Pool Cleanup Demineralizers	4	108	432
Stream Generator Blowdown Demineralizers	4	155	620
Radwaste Demineralizers	2	49	98
Chemical Waste Demineralizer	2	49	98
Polishing Demineralizers	2	49	98
Oil Absorber	2	49	98

Fig. 1. The figure of poly ethylene (PE) HIC

Keywords: Spent Resin, PWR, Characteristics

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PS5 (T5.7-1053)

CSN Regulatory Guide on Methodology for Clearance of Residual Materials Arising from the Operation and Dismantling of Nuclear Facilities

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Nuclear Safety Council (CSN) is the Spanish institution responsible for ensuring nuclear safety and radiation protection of people and the environment. Among other duties, CSN is in charge of the overseeing of clearance processes carried out by licensees of nuclear facilities in Spain. This overseeing activity is supported by the approval of a *Clearance Methodology*, which must meet national requirements and international standards. The fulfillment of this *Clearance Methodology* assures the accomplishment of *Clearance Levels*, and therefore the achievement of the radiological criteria, defined by a maximum dose rate of 10 µSv/year.

The *Clearance Methodology* is developed by licensees, and included in an official document called *"Control Program for Clearable Materials"*, compulsory for dismantling nuclear facilities. For operating facilities, the *Clearance Methodology* supports the residual material clearance authorization, granted by the competent Ministry.

The main purpose of the *Clearance Methodology* is to establish the activities to be conducted by the licensee for the clearance of residual materials. It is a systematic approach to the design and implementation of the clearance processes. Another objective of the *Clearance Methodology* is to ensure the traceability for clearable materials until they reach conventional waste management routes.

The proposed poster presents the ongoing work conducted by CSN on a regulatory Safety Guide about the content of the *Clearance Methodology*. The draft of the mentioned Safety Guide will be submitted for comments to national stakeholders in 2020.

Keywords: *Clearance methodology, Clearance level, Residual material.*

PS5 (T5.7-1084)

Preliminary Evaluation of the Effective Dose Rates of the 2nd Radwaste Receipt & Storage Building

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KORAD is operating the radwaste receipt & storage building with a storage capacity of 7,000 drums for the 1st phase of underground silo. Due to the gradual expansion of disposal facilities, storage and working space are insufficient for the management of radioactive waste. In order to operate the Gyeongju disposal facilities efficiently, the 2nd radwaste receipt & storage building with a storage capacity of 10,000 drums will be constructed. In this study, in order to assessment of the radiation effects of workers, indoor and outdoor exposure doses were evaluated in the 2nd radwaste receipt & storage building. It is assumed that the storage areas in the 2nd radwaste receipt & storage building are full of radioactive waste generated during the operation of the nuclear power plant (NPP). As shown in Figure 1, the effective dose rates were calculated by selecting 3 measurement points for indoor and 4 measurement points for outdoor using MCNP6 code. The total number of drums arranged is 7,856 drums. The source term selected the dry active wastes with the highest specific radioactivity of Co-60 generated from Hanbit NPP among the acquired radioactive wastes by KORAD. As shown in Table 1, the effective dose rates at the outdoor measurement point of the 2nd radwaste receipt & storage building have calculated a maximum of 0.0133 μSv/h. This result is about 1/8 of the effective dose rates of 0.1 μSv/h that can occur from natural radiation sources. It can be seen that the radiation exposure of residents or workers from the outdoor during normal operation is very small. The effective dose rates of the indoor were calculated maximum of 2,078 μSv/h. Because these results were calculated with an excessively conservative source term, it is necessary to be reviewed to reflect the reality of the source term.

Table 1. The effective dose rates of workers (μSv/h)

Measurement point		Effective dose rates (μSv/h)	Measurement point		Effective dose rates (μSv/h)
Indoor	1	1,579	Outdoor	5	0.0096
	2	1,960		6	0.0005
	3	2,078		7	0.0090
	-	-		8	0.0133

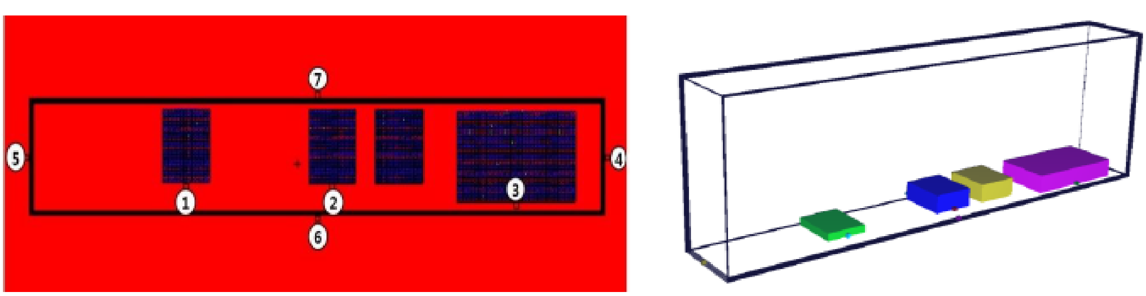

Fig. 1. MCNP modeling of the 2nd radwaste receipt & storage building

Keywords: Radwaste receipt&storage building, MCNP code, Effective dose rates

ACKNOWLEDGMENTS
This research was supported by the Korea Institute of Energy Technology Evaluation and Planning (KETEP), (No. 20181510300870).

PS6 (T5.7-1133)

Regulatory Clearance of Radioactive Soil Wastes

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Uranium contaminated soil waste was generated in KAERI during the renovation of a building that was used as a uranium ore laboratory in the past. Korean nuclear law regulates the acceptance dose criteria for regulatory clearance or radioactivity concentration by radionuclide [1]. The Nuclear Safety Act (NSA) regulates 257 radionuclides in accordance with IAEA RS-G-1.7 [2]. Before regulatory clearance, radiation monitoring, representative sampling and measurement of radioactive concentration was conducted for the soil.

To analyze the activity concentration of the soil waste, representative samples of the soil were collected by 1 ~ 1.5 kg at a ratio of 1 kg to 200 kg. Energy Dispersive X-ray fluorescence(ED-XRF) analysis was used in the measurement of uranium contaminated soil samples. The purpose of the ED-XRF method is to carry out quantitative and qualitative analysis of the elements simultaneously. A total of 340 representative samples were analyzed for 30 minutes per sample by ED-XRF under the analytical condition. For this analysis of the soil, the geochemistry trace powder method for environmental and mineral sample measurement was applied. Radiological assessment for various environmental pathways of the soil landfill disposal was evaluated using the RESidual RADioactivity (RESRAD) code, which is accepted for use by U.S. Department of Energy (DOE), the Nuclear Regulatory Commission (NRC), the U.S Environmental Protection Agency (EPA) and other regulatory agencies. International and domestic reference data were used for the assessment of uranium concentration. The result of exposure dose assessment is shown in Table 1.

Table 1. The result of exposure dose assessment

Exposure dose assessment	
Individual dose, μSv/yr	Collective dose, [man·Sv/yr]
7.77E-01	1.37E-02

Individual and collective dose assessment for clearance as well as the maximum concentration of U-238 should meet permissible criteria. According to the result of its survey in June 2019, a large amount of various clearance level waste including equipment, battery and furniture within the radiation controlled area at KAERI was stored. The clearance of VLLWs from regulation is expected to reduce the amount of radioactive waste and the cost of radioactive waste disposal.

Keywords: *Clearance, Radioactive waste, Dose assessment*

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PS6 (T5.7-1134)

Development of Dose Assessment Program for Worker by Radioactive Waste Clearance Recycling Scenario

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In Korea, it is necessary to satisfy the regulatory requirements of exposure dose for radioactive waste clearance using commercial program. In case of recycling scenario, commercial program does not meet those Korea regulatory guideline due to suspension of update. Therefore, this study developed worker dose assessment program for radioactive waste clearance recycling scenario. For developing the new dose assessment program, we (1) reviewed worker exposure scenario, (2) analyzed and derived the external and internal dose assessment methodology, (3) verified external dose assessment methodology, finally (4) develop the EXCEL-type new program. Development sequence of dose assessment program is given Fig. 1. At first, we reviewed of worker exposure scenario by analyzing recycling process. Finally, we selected target worker and exposure pathway. This study derived external dose assessment methodology. This methodology is composed the radiation concentration formula and dose conversion factor. The radiation concentration formula was derived by incorporating the basic formula for the concentration of radiation of a disk source at the receptor (position P in Fig. 1.) and simplified for convenience of calculation. Dose conversion factor was used to EPA data. To verify newly developed formula, we have compared the results with the commercial program. As a result, the exposure doses of developed equation were found to be similar to results of the Microshield, which is commercial dose assessment program. And Internal dose assessment methodology utilized the methodology of the existing commercial program. Based on the research details, we developed worker dose assessment program for radioactive waste clearance recycling scenario. This research results will be contributed to development of commercial dose assessment program for radioactive waste clearance as a whole.

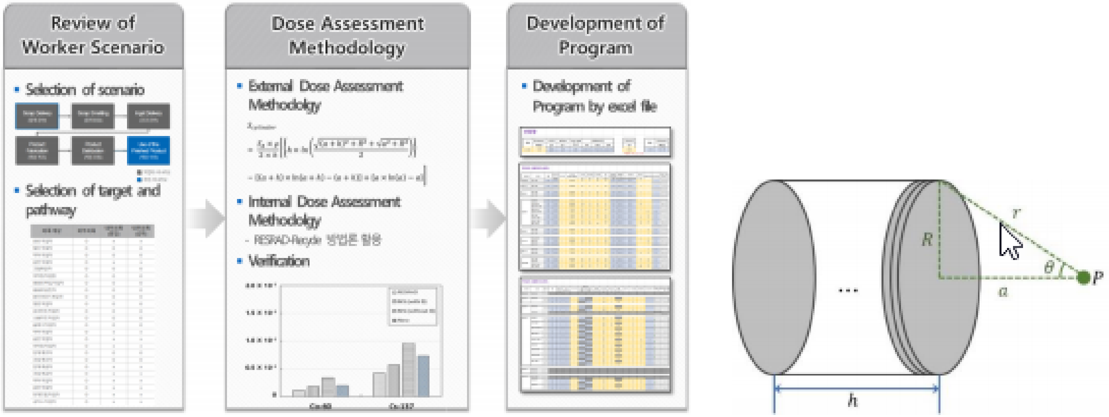

Fig. 1. Development Sequence of Dose Assessment Program and Schematic Diagram for External Dose Assessment Methodology

Keywords: *Radioactive Waste, Clearance, Dose Assessment*

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PS6 (T5.7-1135)

Evaluation of the Maximum Allowable Ratio of Hot Spot Drums for Radioactive Waste Grouping Analysis Using Concentration Averaging Method

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Radioactive Waste Chemical Analysis Center at the Korea Atomic Energy Research Institute (KAERI) has used the grouping analysis that groups waste samples with identical characteristics for efficient analysis of contaminated trash (e.g. combustible waste, HEPA filter). However, it is currently not easy to apply the grouping method due to the difficulty of decision on optimum amount of drum for grouping and proving its validity. To solve these issues, we reviewed the concentration averaging method proposed by US Nuclear Regulatory Commission (NRC) and evaluated the maximum allowable ratio of hot spot drums which have relatively high radioactivity when grouping waste samples. The evaluation process is as follows.

First, we determined the radionuclide inventories of reference waste drum and hot spot drum. The radionuclide inventories of reference waste drum were assessed based on maximum values for each nuclide among the radionuclide concentrations analyzed for disposal from 2015-2018. The radionuclide inventories of hot spot drum were set to the same value as the disposal concentration limits for the most conservative situation. Second, we performed waste classification according to the method proposed by 10CFR part 61.55 [1]. As a result of the waste classification, the reference waste drum was classified as "Class C waste" and the hot spot drum slightly exceeded class C limit. Third, we set up the hot spot inflow scenario and evaluated the allowable number of hot spot drums when grouping the waste samples. The hypothetical situation is the continuous grouping of a certain number of hot spot drums into the reference drums. And then we evaluated the proportion of hot spot drums to the total number of drums grouped. For this evaluation, we reviewed the concentration averaging method proposed by US NRC report [2]. This report provides a volume threshold that allows averaging of waste concentration, and a method for assessing whether the threshold is met based on the class of waste mixture and certain factor (i.e. sum of fractions). We applied this concentration averaging range as an allowable grouping range, and evaluated how many hot spot drums could be included within the allowable grouping range. As a result of this evaluation, maximum allowable ratio of hot spot drums when grouping certain number of drums is shown in table 1.

Table 1. Maximum allowable hot spot drums according to the number of drums grouped (part of the results)

Number of drums grouped(200L)	Maximum allowable hot spot drum quantity (200L)	Ratio of hot spot drums(%)
3	1	33.3
4	1	25
	……	
26	9	34.6
	……	
49	1	2

Based on this evaluation, we were able to identify the optimally acceptable hot spot quantity for each grouping drum quantity. By developing this process and further study, we expect that this methodology will be adapted to establish an optimal grouping range that can be used to the analysis of KAERI's radioactive waste.

Keywords: *Concentration averaging, Hot spot drums, Grouping analysis*

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PS5 (T5.7-1146)

Microbial Diversity of Preselected Bentonite and Beishan deep Groundwater in DGRs

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High-level nuclear waste (HLW) stored at long-term stage in deep geological repositories(DGRs) is an option favored by many countries, a better knowledge of waste container corrosion and radionuclide migration in repository conditions plays an important role in safety assessment.Microbial activity in groundwater and Bentonite may significantly influence the function of future deep repositories, mainly because of microbe-induced corrosion (MIC) and radionuclide migration.This study focused on investigating the microbial communities of deep groundwater in Beishan and Gaomiaozi bentonite, based on Illumina sequencing and 16S rRNA analysis.The results showed a high bacterial diversity in Groundwater and the bentonite samples, dominated by Archaea and abundant other bacteria. The community of Archaea included Nitrososphaera, Thermogymnomonas, and Methanobrevibacter,The bacteria was dominated by firmicutes and proteobacteria,which exist mainly in the form of spore, and therefore the use of bentonite as a barrier material should be carefully considered.

Keywords: *High-level nuclear waste; deep geological repositories; Microbial Diversity; Bentonite; Groundwater*

PS5 (T5.7-1157)

CSN Regulatory Guide on Methodology for Clearance of Residual Materials Arising from the Operation and Dismantling of Nuclear Facilities

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The purpose of this study is to perform regulatory clearance for recycling of 200L drum used for radioactive waste storage. Expected contamination nuclides were selected U-234, U-235 and U-238 by the history of use. In order to ensure the representativeness of sampling issue for radioactivity analysis, it was cut in various directions as shown in Fig.1. In this process, producted 5 mm size pieces were collected, it is to minimize porosity in the sample container. Measurement of a gamma-emitting radionuclide, Pa-234m, using HPGe (High Purity Germanium) detectors was implemented to analyze concentration of U-238 which is the parent nuclide of that indirectly. Concentrations of U-234 and U-235 were calculated based on the nuclear characteristics and radioactivity fraction with U-238. Also, we tried to prove non-contaminations from the another radionuclides with the gamma-ray spectrometry and additional analyses for gross-alpha and -beta. The collective dose was verified that were below the regulatory condition under the dose assessment using the steel recycling scenario of RESRAD-RECYCLE 3.10 (Argonne National Laboratory, USA). We consider that, in the future, our method can be utilized to regulatory clearance for a number of used radioactive waste drums.

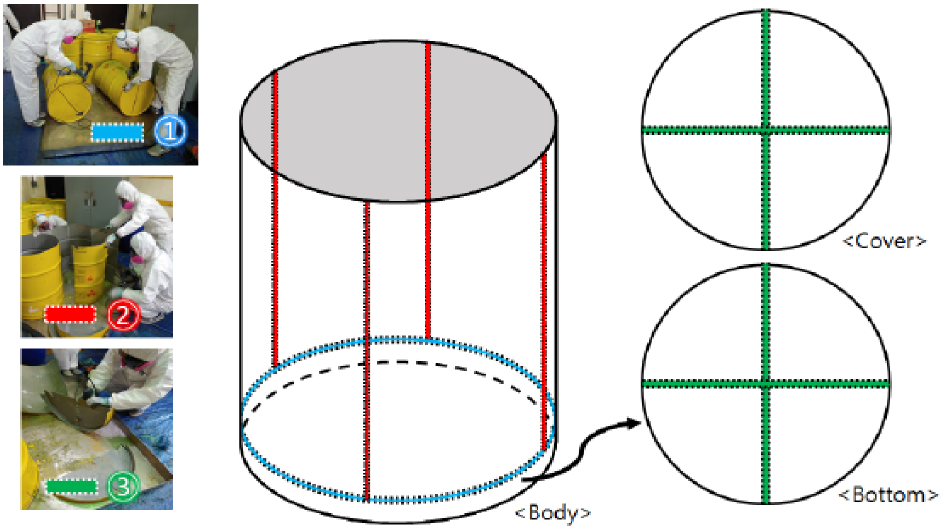

Fig. 1. Sampling procedure for radioactivity analysis

Keywords: Reguratory Clearance, Radioactive waste drum, Sampling

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PS5 (T5.7-1160)

Radiological Risk Assessment due to Naturally Occurring Radioactive Material (NORM) in the Proposed Radioactive Waste Storage Area of Pilikwe, Botswana

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Determination of NORM concentrations followed by the radiological risk assessment were conducted and then characterised based on the activities of the identified nuclides for the different stages of the radioactive waste storage facility to be constructed at Pilikwe village of central district in Botswana. Sixty soil and two borehole water samples were collected representing the area of construction site and its surrounding. The radioactivity measurement was performed using high-resolution gamma-ray spectrometry in a low background configuration at the Centre for applied radiation science and technology laboratories, North west university. The attained activity concentration results for the soil samples show that max value for Ra-226 is 36.308 Bq/kg, min value is 6.658 Bq/kg and average value is 15.853 Bq/kg; max value of U-238 is 53.471Bq/kg, min value is 7.754Bq/kg and average value is 25.550Bq/kg; max value of Th-234 is 46.782Bq/kg, min value is 11.580Bq/kg and average value is 24.976Bq/kg and the max value for K-40 is 468.600Bq/kg, min value is 59.150Bq/kg and average value is 201.304Bq/kg for the Pilikwe area before any construction work of the radioactive waste storage facility is conducted.

The average estimated absorbed dose rate (D) for soil samples was 35.289 ± 2.261 nGy/h, which was lower than the worldwide absorbed dose rate of 57 nGy/h for soil. The average estimated annual effective dose equivalent (AEDE) from soil samples was 43.2782 ± 2.773 µSv/y, which was lower than the recommended worldwide value of 70 µSv/y for soils. The radium equivalent activity (Ra_{eq}) for soil was 76.775 ± 5.0509 Bq/kg. The external hazard index (H_{ext}) for soil was 0.207362 ± 0.014. The mean Ra_{eq} values for soil, bottom ash, coal and fly ash were all below the worldwide accepted value of 370 Bq/kg. The average H_{ext} value for soil, was found to be below the worldwide recommended value of one. All the hazard indices show that the samples from Pilikwe village area and surrounding have acceptable indices with no hazard. Thus, a radioactive storage facility can be built in the identified location provided there is proper management and safeguarding the leaching of the stored radiative into waste into the soil, as well as having measures in place for monitoring and inspecting the NORM radioactivity concentrations of the area on specified time frames for assuring that there are no nonconformities

***Keywords**: NORM, radiological risk assessment, gamma spectrometry.*

ACKNOWLEDGMENTS
Special thanks go to my supervisors Prof. Manny Mathuthu and Dr. Chamunorwa Oscar Kureba, North West University, International atomic energy agency (IAEA) and Radiation protection inspectorate.

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PS5 (T5.7-1182)

NORM Management in Brazilian Scrap Metal

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NORM is often found in steel pipes used in oil and drilling operations from O&G industry. The incrusted material, which contains small amounts of radium (Ra), is accumulated in metallic pipes or equipment. When the metal is discarded without proper care, it can end up in a steel mill. The radioactive material associated with scrap metal can cause negative health effects, as well as economic problems for the steel industry and public acceptance issues. This problem came up a few years ago and concerns have been expressed by the metal recycling and production industry. In Brazil, the normative responsibility on the NORM subject belongs to CNEN - National Commission of Nuclear Energy. In addition to not having an effective system to control the radioactive scrap, Brazil also has another problem: there is no definitive solution for disposal of NORM waste that is currently accumulated in temporary deposits of steel mills that receive this type of material. The study aims to analyze the current reality of Brazilian steel mills that receive pieces contaminated with NORM and also suggests alternatives for managing the material (Table 1). Therefore, radiochemical analysis and radiometric monitoring were performed on contaminated scrap metal received by Brazilian facilities of 2 (two) different large companies, in order to obtain information and suggest a Brazilian standard for radioactive scrap management. Thus, management alternatives were raised. The results of the radiometric monitoring on 8 (eight) selected pieces of scrap metal varied from 1.37 to 20.0 $\mu Sv.h^{-1}$ and the radiochemical analysis carried out in the laboratory through gamma spectrometry showed results that varied from 22.75 to 79.7 $kBq.kg^{-1}$ for Ra-226. These values are above the limit established for disposal of solid waste in Brazil and show the urgency to establish a national procedure so that steel companies can deal with this issue safely.

Table 1. Survey of alternatives for NORM waste management.

Alternatives	Estimated Cost	Estimated Time	Required Approvals	Third Part Services	Viability
Return to origin (shipment rejected, without segregating radioactive material)*	$$	-	Internal shipment rejection procedure	N / A	★★★
Return to origin (with segregation of radioactive material)	$$$	◷◷	CNEN, Environmental Agencies and the scrap supplier	Transport Service	★★★
Scrap control at the supplier	$$	-	Agreement between mill and scrap supplier.	N / A	★★★★
Decontamination of scrap metal containing NORM	$$$$	Depends on the volume	CNEN and Environmental Agencies	Radiation Protection and Transport Services	★★
Export of NORM waste	$$$$$	◷◷◷	CNEN and Environmental Agencies	Radiation Protection, Transport and Final Disposal Services	★

***Keywords**: NORM, Scrap Metal.*

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PS5 (T5.7-1194)

Preliminary Study on Passive Neutron Emission Tomography for Monitoring PWR Spent Fuels in Dry Storage

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Establishing the appropriate safeguards approach to resolve the challenges associated with verifying spent fuel assemblies stored in dry casks has been one of the major priorities for the IAEA and the Member States in the past decade. In particular, verifying the declared contents in dry casks through a non-intrusive technique is fundamentally challenging due to the physical barriers of a dry cask such as concrete walls, metal walls, and neutron absorbers. Although several techniques have been proposed to satisfy the safeguards requirements, no reliable methods have yet been established for inspecting the integrity of spent fuels in dry casks.

Recently, KINAC developed a prototype safeguards system that utilizes a passive scanning to obtain the cross-sectional image of spent fuel assemblies in dry casks. The prototype consists of He-4 gas scintillation neutron detectors, collimators, and an integrated control system designed for the detector operation and signal processing. The prototype system was tested a 1/10 scaled-down TN-32 cask fabricated in collaboration with a manufacturing company and Cf-252 neutron sources. Also, the obtained experimental results were inter-compared with Monte Carlo simulation results to evaluate its technical feasibility. Furthermore, various image reconstruction algorithms based on iterative methods were evaluated using the acquired data to examine their performance in terms of the quality of the resultant images. Through image comparison, we determined that the simultaneous iterative reconstruction technique (SIRT) provided the best image quality with the designed prototype system.

This work investigates the ability of the new prototype safeguards system to detect removal or substitution of spent fuel assemblies in dry cask storage. Furthermore, the detection of various source configurations is explored to optimize the designed system.

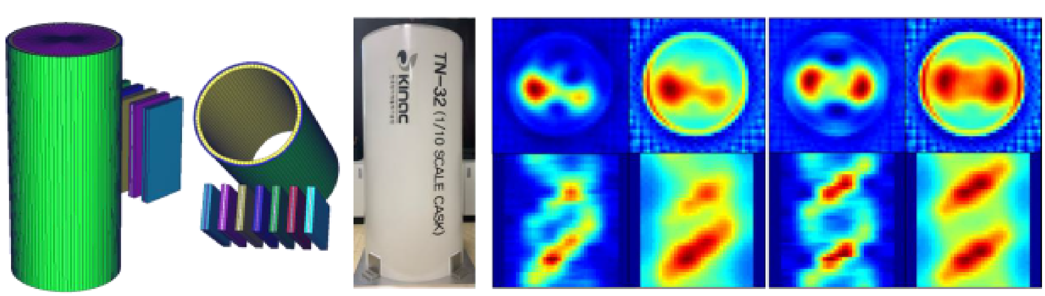

Fig. 1. The lab-scale prototype system for imaging internal structures of a dry cask

Keywords: *Dry Storage, Safeguards, Neutron Tomography*

ACKNOWLEDGMENTS
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PS5 (T5.7-1244)

Derivation of Conversion Factor for Radioactivity Concentration to Surface Count Rate of Natural Uranium-contaminated Soil Using Microshield

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In nuclear facilities, large amounts of waste are classified as radioactive waste during sampling and classification although clearance is possible. This in turn increases the cost of treating radioactive waste. If a primary classification system using surface count rate can be applied during the waste sampling process, it will be of benefit in reducing waste disposal costs. The Korea Atomic Energy Research Institute has a huge amounts of soil waste generated by natural uranium conversion facility dicommisioning in 2010. The purpose of this study is to derive a conversion factor that estimates the radioactivity concentration from the surface count rate of this soil waste using computer simulation program microshield. In this soil, various radionuclides exist but according to the waste stream from the facility handling natural uranium, only U-234,235,238 are contaminated except natural origin radionuclides. In order to calculate the count rate-radioactivity conversion factor, the dose contribution by nuclide is required. For this soil, the radioactive contribution of natural uranium nuclide(U-234,235,238) was applied and using microshield, the surface count rate generated when 1Bq/g of natural uranium(U-234,235,238) was estimated. This calculation was made on the assumption that natural uranium maintains its natural proportion. Thereafter, the actual counting rate was measured through the soil sample for which the concentration analysis was completed to confirm the validity of the microshield result. All actual measurements were performed under the same conditions with the computational simulations. The measuring position of the detector was set to (1).1cm and (2).10cm above the surface, to consider the off-centered effect caused by hot spots in the waste. Through this study, it was confirmed that if the contamination source information and the dose contribution of each nuclide in radioactive waste were known, a certain level of correlation between the count rate and radioactive concentration could be identified.

Table 1. Surface count rate compare with using microshield and measurement

No.	Activity [Bq/g]	Above 1cm			Above 10cm		
		microshield	Measured	m/M ratio	microshield	Measured	m/M ratio
1	1.83E+01	91	88	1.03	45	30	1.50
2	2.21E+01	110	150	0.73	54	85	0.64
3	5.22E+01	259	343	0.76	128	163	0.79
4	1.05E+02	524	784	0.67	258	301	0.86
5	1.36E+02	674	950	0.71	332	523	0.63
6	1.67E+02	831	860	0.97	409	552	0.74

***Keywords**: Surface count rate, Radioactivity concentration, Radioactive waste*

PS5 (T5.B-0473)

Radiophobia in Brazil: Causes, Reactions and Consequences

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Lately we have witnessed a decrease in industrial radiology activities in Brazil, especially industrial radiography, especially in the last 20 years. This decrease has causes such as the advancement of manufacturing and engineering technologies, the improvement of quality systems, but also with a silent component that has been increasing every year.

The radiophobia. Unjustified fear related to industrial radiography activities. Year after year there has been a growing prejudice against professionals and companies employing radiological techniques in the labor market, resulting in unemployment of operators and radiation protection officers, reduction of companies with this activity, reducing the operational reliability of energy companies and materials manufacturers. millions of dollars a year to these same companies with unscheduled production shutdowns.

We can establish to the accidents of Chernobyl and Goiânia, the most attentive and critical look of society to the theme of the use of ionizing radiation in the civil sphere. This paper analyzes the causes and consequences of radiophobia in the area of industrial radiography in Brazil.

***Keywords**: Radiophobia, industrial radiography*

PS6 (T6.1-0249)

Development of Operational Intervention Levels for Events with Release of Radionuclides

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An essential part of a protection strategy for radiological emergencies is the development of national dose criteria and of operational intervention levels to decide about protective measures for all 10 scenarios Germany is preparing for. For the process of planning and implementing such protection strategies as required by the German Radiation Protection Law the Federal Ministry BMU has commissioned the German Radiation Protection Commission (SSK) to recommend dose criteria and operational intervention levels (OILs) for emergency response measures.

The choice of a dose criterion for a specific protective measure should be compatible with the overriding reference value – for Germany 100 mSv residual effective dose in the first year – and also achieve that the implemented emergency response measures are justified and commensurate to do more good than harm. Operational intervention levels are needed to link a chosen dose criterion for a protective action with a suitable measurement of the contamination situation such as ambient dose rate (µS h^{-1}), contamination level on surfaces (Bq cm^{-2}) or activity content (Bq g^{-1}, Bq cm^{-3}). This link should adequately model the exposure of persons during a defined exposure period (e.g. 7 days, one year) caused by the measured contamination. Dose calculations to quantify OILs should address the representative person and apply assumptions and parameter values that are in tendency realistic and not unduly conservative.

Operational interventions levels have been developed for the following emergency response actions based on radiation measurements:

- Sheltering on the basis of dose rate (µSv h^{-1}) and contamination level (Bq cm^{-2})
- Evacuation on the basis of dose rate (µSv h^{-1}) and contamination level (Bq cm^{-2})
- Establishing a radiological hazard area to implement access and contamination control on the basis of dose rate (µSv h^{-1}) and contamination level (Bq cm^{-2})
- Contamination control and possibly decontamination of persons and objects (items, goods, vehicles, etc.) based on contamination level (Bq cm^{-2})
- A set of precautionary early actions: warning of population to consume freshly contaminated food and agricultural measures to reduce food contamination based on dose rate (µSv h^{-1})
- Application of maximum permitted levels of radioactive contamination of food and feed (Bq kg^{-1}) according to Euratom Regulation

An OIL value for the protective measure "iodine thyroid blocking" on the basis of measurement of contamination levels has not been proposed because to be effective it should be based on early warning, preferentially before release from a nuclear reactor. Also for the decision of temporary or long-term relocation of affected population an OIL has not been proposed. As dose criterion the reference value of the residual effective dose in the first year would be applied. Being not an urgent measure within a few days such a serious decision would have to be based on a cautious balancing of health risks from exposure and the heavy burden for the population.

Keywords: Emergency response measures, operational intervention levels

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PS6 (T6.1-0561)

Investigation on the Research Trends using X-ray for Nuclear Forensics and Preliminary Experiments by Measuring Uranium Samples in Portable XRF

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According to the IAEA's Incidents and Illegal Transactions Database (ITDB), nuclear and radioactive materials that have consistently been reported stolen or lost since the collapse of the Soviet Union in 1991. These materials pose a potential threat to the general public as well as to vulnerable national infrastructure. The international community has held several summit meetings to determine appropriate, active responses to nuclear terrorism exploiting these materials, and has emphasized the importance of nuclear recognition as one of the primary means for nuclear security.

In response to such objective, KINAC has completed initial research to establish a national nuclear forensics system. Currently, further research is underway to advance sample analysis and measurement techniques applicable to nuclear forensics. In this study, we investigated research trends and analysis techniques that utilize x-rays as related to nuclear forensics. In addition to this, we studied preliminary experiments and the analysis results of measurement spectra to be conducted for the possible application of nuclear forensic of portable XRF that can be used for the elemental/chemical characterization of nuclear material in the event of discovery of unknown nuclear material.

PS6 (T6.1-0716)

A Study on Reasonable and Efficient Pre-distributing Methodology of KI Applicable to the Republic of Korea

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Taking KI is a measure for radiation protection in the event of a nuclear emergency. Therefore, national plan of republic of Korea is to protect local residents in nuclear facilities by stockpiling KI in peace times and distributing them in case of nuclear emergency. However, the radiation protection effect can be different depending on the time difference between points of taking KI from point of exposure to radioactive iodine. Therefore, many countries have EPZ (Emergency Preparedness Zone) from NPP (Nuclear Power Plant) plan to protect residents in EPZ by pre-distributing KI. However, because each country has different social cultural backgrounds and policy directions, different 'pre-distributing KI methodology' is adopted. Republic of Korea is a nation that utilizes 24 NPP and 1 research reactor. Therefore, Koreans are interested in various measures that can minimize the risks that may arise from nuclear facilities. In particular, after the Fukushima nuclear power plant accident in neighboring Japan, people became more interested in radiological protection measures. Moreover, in the event of a complex disaster situation, there is a possibility that KI may not be distributed to residents at the appropriate time due to various variables from accident. In addition, in the event of a complex disaster situation, time for evacuation measures can be longer than expected, so it is important to take KI at the appropriate time to ensure maximum radiation protection. In order to solve these problems, this study developed a rational and efficient 'pre-distributing KI methodology' that considers the social cultural background and policy direction of Republic of Korea. The 'pre-distributing KI methodology' developed in this study consists of 4 steps. Details are shown in fig 1.

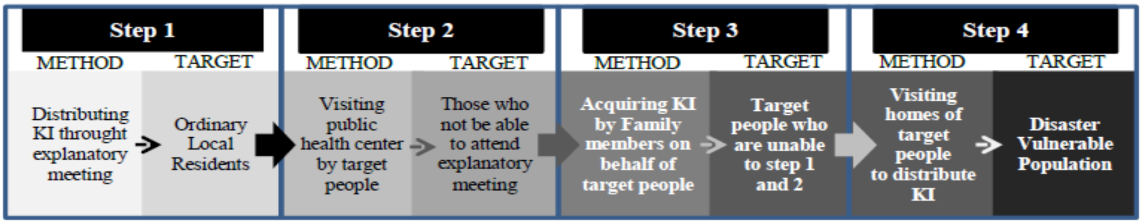

Fig. 1. Four-Step Pre-Distributing KI Methodology

In addition, it proposed a revision of the current legislation in order to implement developed 'pre-distributing KI methodology. The revision of the legislation focuses on 'Act on Physical Protection and Radiological Emergency', but in order to minimize the problem of conflicting legal interpretations between the two laws, the relevant laws such as 'Pharmaceutical Affairs Act' were intended to be revised together.

***Keywords**: Pre-distributing KI Methodology, Nuclear Emergency, Radiation Protection Measure*

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PS6 (T6.1-0789)

Development and Implementation of Protection Strategies in Nuclear and Radiological Emergencies in Nigeria: Challenges and Lessons Learned

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In 2015, the International Atomic Energy Agency (IAEA) published a new standard on Emergency Preparedness and Response, GSR-Part 7 which contained a requirement on protection strategies (Requirement 5) in nuclear and radiological emergencies. According to the requirement, the government shall ensure that protection strategies are developed, justified and optimized at the preparedness stage for taking protective actions and other response actions effectively in a nuclear or radiological emergency. Nigeria is not a nuclear power country but has the intention to operate a nuclear power plant in the near future. However, one of the key elements of emergency preparedness and response for both nuclear power countries and the embarking countries like Nigeria is developing national generic criteria in terms of projected dose or of received dose for taking protective actions and other response actions (either individually or in combination) and establishing pre-established operational criteria (i.e. observable conditions on the site, emergency action levels (EALs) and operational intervention levels (OILs)) on the basis of national generic criteria for initiating the different parts of an emergency plan and for initiating protective actions and other response actions. This paper highlights the steps involved in developing and implementing protection strategies based on the requirements of GSR Part 7 and the challenges faced by Nigeria in implementing Requirement 5 of GSR-Part 7 on protection strategies and to share lessons learned.

PS6 (T6.1-0852)

The Public Health Center Roles during Radiation Emergency in Mongolia

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Mongolia is a landlocked country in Central Asia and East Asia with large area approximately 1,564,116 square kilometers and a population of 3.2 million inhabitants. Mongolia located between China and Russia. Mongolia is a non-nuclear country, since there are no nuclear power plants and research reactors. But Radiation sources and radioactive materials are being used quite well for the following sectors of the country such as Medicine (radio diagnostics and treatment) Animal husbandry and Agriculture, Industry, Geology and Mining, Science and Education, Natural Environment. The Nuclear Energy Agency is one of the competent authority of the and under the auspices of the Government is responsible for development of policy for the activities relating to development of nuclear research and technology, radiation protection and safety. The law of Mongolia on Nuclear Energy has been enacted July 2009 and its purpose has been described as to regulate relations pertaining to exploitation of radioactive materials and nuclear energy on the territory of Mongolia for peaceful purposes, ensuring nuclear and radiation safety and protecting population society and environment from negative impact of ionizing radiation. The NEA has been pointed out several priorities areas for developing radiation safety and security since its establishment. One of the major stressing points were always been strengthening and capacity building on Radiological Emergency. In 2017, The Regulation on Information exchange between sectors and rapid response during potential Disasters and Public Health Emergencies has been approved by Deputy Prime Minister who is a Chair of State Emergency Commission. The objective of this regulation is to ensure preparedness and rapid response and to coordinate multi-sectorial collaboration through the provision of accurate information to decision makers based on joint risk assessment and exchange information about an event, which could potentially lead to Public Health Emergency. The radiation Emergency itself considered to be the main aspects of Public Health Emergency. This order also has been described as who is and are a lead and collaborating organizations during PHE. The General Authority for the Specialized Inspection has been appointed as a leading body during Radiation incident and poisoning. The NCPH also appointed as one of the main collaborating organizations for the Radiation Emergencies. Our main function is to rapid assessment of Health and Medical needs and delivery, Community needs, Evacuation issues, Injury and illness Surveillance, Registration, Pharmaceutical supply, Food, Shelter and Potable water, sanitation, waste management, Risk communication, Rumor control, Mental health, Vector control, Handling deceased during Radiological Emergencies.

Keywords; Nuclear Energy Agency1, General Authority for the Specialized Inspection2, National Center for Public Health3

PS6 (T6.1-0898)

Monitoring of Radionuclides in the Atmosphere by German Meteorological Service– Atmosphere Transport Dispersion Models and Measurements

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Based on national laws and ordinances the German Meteorological Service (Deutscher Wetterdienst, DWD) is operating measuring systems for the fast detection of aerosol bound radionuclides at measuring sites evenly distributed over the territory of Germany. Naturally occurring and manmade radionuclides in the atmosphere are measured continuously at 48 measuring sites. The data are summarized at the headquarters of DWD and forwarded to the Federal Office of Radiation Protection (BfS). A summary of the results is made public on the internet site and in the yearly report of the Ministry for Environment, Nature Protection and Nuclear Safety (BMU). The monitoring systems and the radiochemical separation procedures used in the laboratory are presented, including measurements of the gamma emitting radionuclides, total alpha- and total beta-activity, tritium in precipitation, isotopes of uranium, plutonium and americium as well as Sr-89/90 and the noble gases Kr-85 and Xe-133.
In case of an accident with the release of radionuclides into the atmosphere dispersion calculations give valuable information about the expected transport way of contaminated air. Measuring data will be transferred in intervals of two hours. Samples (filter, precipitation) are transported to the radiochemical laboratory.
With the information about the activity concentration of the radionuclides in the air, in precipitation and on the ground, first measures to avoid the exposition of high radiation is recommended by the Ministry.

Table 1. Monitoring of radionuclides in the atmosphere in the emergency mode

Monitoring systems	Number of measuring sites	Interval of data transfer	Limit of detection
gamma emitting aerosol bound radionuclides	41	2h	1 Bq/m³
Aerosol bound artificial alpha activity	48	2h	0,5 Bq/m³
Aerosol bound artificial beta-activity	48	2h	1 Bq/m³
Gaseous iodine isotopes	48	2h	1 Bq/m³
Gamma emitting radionuclides on the ground	38	2h	1500 Bq/m²
Gamma emitting radionuclides in precipitation	40	daily	5 Bq/L
Xe-133	1	daily	10 kBq/m³

Keywords: *airborne radionuclides, ATDM, monitoring of radioactivity in the atmosphere*

ACKNOWLEDGMENTS
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PS6 (T6.1-0910)

Real Time Tracking of Mobile Radioactive Sources

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It is vital that mobile radioactive sources remain under safe and secure control, at all times and the operator is notified in real time of their unauthorised removal from a fixed location, store or vehicle. This paper describes a system that constantly monitors mobile radioactive sources and provides a means to detect their location should unauthorised removal occur. Thousands of radiographic exposure devices are transported in the public domain every day and there are examples where the vehicle carrying a device has been stolen, not for the device itself but for the vehicle, which often leads to the device being discarded. Abandoning the device can lead to high risks of exposure for those who have secondary contact for example children or those attracted by its perceived intrinsic value. A lack of knowledge of the dangers of radiation has led some people in the past attempting and sometimes succeeding to open a device and remove the source. With other factors involved such as location, country and the intentions of those in possession of the device, it is therefore important that the device is located quickly, enabling recovery actions to begin as soon as possible. In addition, this paper encourages that such monitoring systems are more widely promoted in all countries and that an international code of practice is developed to assist when sources are transported internationally.

PS6 (T6.1-0982)

Lucas Heights Radiological Hazard Assessment and Protection Strategy

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The Australian Nuclear Science and Technology Organisation (ANSTO) has a campus at Lucas Heights, New South Wales, where it operates a modern research reactor, manufactures nuclear medicines for the Australian and world markets and carries out extensive research into many diverse fields. It is a unique radiological and nuclear site within Australia and, as such, it is imperative that the local authorities and emergency response services have a clear understanding of the potential hazards that could be caused in a major accident at the facility and the appropriate responses. The Lucas Heights Radiological Hazard Assessment and Protection Strategy has been prepared as a technical document in support of the Lucas Heights subplan to the NSW State Emergency Management Plan. This details the hazard assessment and protection strategy for aspects of the preparation for, response to, and immediate recovery from a radiological or nuclear emergency occurring at Lucas Heights. It has been prepared on behalf of the State Emergency Operations Controller and was endorsed by the State Emergency Management Committee. It recognises changes in the facilities at the Lucas Heights Science and Technology Centre as well as a revised hazard and threat environment. It also revises recent changes in emergency arrangements and international standards. It supports the preparation for and the response to a substantial radioactive release, including radiological monitoring, other response actions, and communication with relevant parties.

This poster summarises the potential combined off-site radiological impacts of reference accidents for the most potentially radiological hazardous buildings at Lucas Heights, and the protection strategy that would be employed if such a combined scenario were to occur. For the LHRHAPS the ANSTO Lucas Heights campus fence line is prudently considered the boundary between on-site and off-site in line with International Atomic Energy Agency (IAEA), Preparedness and Response for a Nuclear or Radiological Emergency, IAEA Safety Standards Series No. GSR Part 7 (2015).

The off-site radiological impact of the combined reference accidents is assessed for adults, children and infants over an exposure period of 50 years using conservative assumptions, such as unfavourable meteorological conditions and lack of response actions, for example no sheltering during the exposure period.

Projected effective doses are shown to be below generic intervention levels (GILs) requiring evacuation as an urgent protective action for all population groups. GILs for sheltering would be exceeded for all population groups within 550 m from the Lucas Heights reference point (the old HIFAR reactor building). GILs for iodine thyroid blocking would be exceeded for adults, children and infants within 150 m, 850 m, and 1100 m, respectively, from the Lucas Heights reference point.

This document was written by a Technical Sub Group of a Sub Committee authorised by the NSW State Emergency Management Committee (EMC) and Sydney Metropolitan Regional EMC in a cooperative arrangement involving Sutherland Shire Council (Chair and Secretariat), Sutherland Shire LEOCON, NSW Police Force (Sutherland Shire PAC, Sydney Metropolitan REMO, Emergency Management Unit and Police Media), Fire and Rescue NSW (Zone Commander and State HAZMAT/CBRN Sub Group member), Rural Fire Service, ANSTO, ARPANSA, NSW Health (Local Emergency, Health Emergency Management Unit and Environmental Health Branch), NSW Ambulance, Engineering (SEMC representation), Environment Protection Authority, State Emergency Service, Emergency Information Coordination Unit, Department of Primary Industry and Department of Education.

Keywords*: Hazard Assessment, Protection Strategy, Emergency Management*

PS6 (T6.1-1154)

Emergency Preparedness and Response Plan for Pilot Nuclear Fuel Cycle Facility (PNFCF)

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Safety features in Nuclear Fuel cycle facilities are the most important parameter that determines if the public will accept it or not. Whenever public when heard about the word Nuclear the only thing they could imagine is destruction or radioactive hazards on human health. Nuclear safety regulations have been developed and matured in the last decades in a very impressive way, the risk of accidents in nuclear Fuel cycle facilities is low and declining. In over 16,000 cumulative reactor-years of commercial operation in 32 countries, there have been only three major accidents to nuclear power plants: Three Mile Island (USA 1979) where the reactor was severely damaged but radiation was contained and there were no adverse health or environmental consequences, but another two accidents that have caused a great release of radioactive material and have required large scale evacuation of contaminated areas are Chernobyl (Ukraine 1986) where the destruction of the reactor by steam explosion and fire killed 31 people and had significant health and environmental consequences. The death toll has since increased to about 56, and Fukushima (Japan 2011) where three old reactors (together with a fourth) were written off and the effects of loss of cooling due to a huge tsunami were inadequately contained. This paper presents how to decrease risk or their consequences for pilot nuclear Fuel cycle facility (PNFCF), prevent severe direct health damages, and reduce the risk for occurrence probabilities of stochastic influence on health damages in the rate as it is reasonably reachable. Considering the identified importance of the EPR for an pilot nuclear Fuel cycle facility (PNFCF) that focuses on main parts of On-site and Off-site emergency, including organization, response roles responsibilities, drills and training, and classification of events if it is alert (1st degree) which including actual or potential substantial degradation of the safety level, the shift supervisor would classify the event, Call Emergency Team, Emergency Response Organization (CE, ERO), and Notify Regulatory body, as well as the officer of radiation safety, would dispatch monitoring team and evaluate the doses. If it is an On-site emergency (2nd degree) which leads to radioactive material release outside of buildings of the nuclear facility or an Off-site emergency (3rd Degree) that leads to serious radioactive material release to the surroundings of the nuclear Fuel cycle pilot facility (NFCPF), the shift supervisor would classify the event, call Emergency Team, notify ERO, Regulatory body and local authorities, also the officer of radiation safety would take Protective measures dispatch monitoring team and evaluation of doses and the Mobile monitoring group would surround the plant for Evaluation of Radiation consequences.

PS6 (T6.1-1169)

Penetration Factor and Indoor Deposition Rate of Elementary and Particulate Iodine in a Japanese House for Assessing the Effectiveness of Sheltering for Radiation Exposures

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Sheltering is one of the protection actions against radiation exposures in a nuclear accident. The effectiveness of sheltering is often expressed by the ratio of the indoor to the outdoor cumulative radioactivity concentrations or doses. The indoor concentration is mainly controlled by the air exchange rate, penetration factor, and indoor deposition rate. The penetration factor is defined as the ratio of the nuclides that penetrate indoors from the outdoors. The indoor deposition rate is defined as the rate of the nuclides removed from indoor air because of the deposition on floors, walls, and ceilings

For iodine, which is one of the most important nuclides for assessing the dose to the public, the penetration factor and indoor deposition rate differ among chemical forms (i.e. elementary iodine (I_2), organic iodine, and particulate iodine). However, especially for I_2, there is few data on the penetration factor and indoor deposition rate. Additionally, these parameters of particulate forms differ depending on materials of the wall and opening part. Therefore, it is not proper way to apply these parameters obtained in some countries directly to other countries' houses.

In this study, we investigated the penetration factor and indoor deposition rate of I_2 and particles by two experiments: (1) the "house experiment" conducted in real Japanese houses and (2) the "laboratory experiment' conducted with a chamber which can control various environmental factors (i.e. temperature, humidity, and air exchange rate). The experiment for I_2 was only conducted in laboratory experiments because of its toxicity. The house experiment was performed in two apartment houses and three single-family houses. The laboratory experiment was performed in a chamber (35 cm × 70 cm × 35 cm) which was separated in the middle using an opening simulating a ventilation opening.

The penetration factor ranged 0.5–1 for particles of 0.3–1 μm and 0.2–0.6 for I_2 depending on the air exchange rate. The indoor deposition rate for a house room ranged 0.01–0.3 h^{-1} for particles of 0.3–1 μm and 0.5–2 h^{-1} for I_2 depending on the floor material (tatami, wood, carpet) and the amount of furniture. The indoor deposition rate of particles obtained by the house experiments was approximately eight times smaller than that obtained by laboratory experiments due to the difference of surface-to-volume ratio between a house room and chamber. Therefore, to evaluate the deposition rate for a room, the indoor deposition rate of I_2 obtained by the chamber experiments was corrected by taking into account the difference of surface-to-volume ratio between the chamber and the room.

Keywords: *Penetration factor, Indoor deposition rate, Sheltering, Radiation exposure, Radiation protection*

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PS6 (T6.2-0036)

On the Use of Field Experiments to Derive First Responder Guidelines to Dealing with RDD-incidents

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Although access to and use of radioactive sources is strictly regulated, one may never rule out the possibility that disaffected groups may gain access and weaponize them to cause panic, mass disruption in addition to inflicting huge financial losses associated with the site recovery operations.

Under the 7th framework program the European Commission co-funded the CBRN crisis management, Architectures, Technologies and Operational procedures (CATO) project. CATO was meant to deliver a fully customizable decision support system for managing CBRN related incidents, in addition to developing incident commander and first responder guidelines.

During the project execution a set of controlled field experiments was carried out with the purpose of assessing and evaluating the performance of the decision support system during a real event, as well as, generating the necessary data and information to derive practical guidelines for first responders in the event of a radiological dispersal device (RDD) incident. The main difference between our experiments and those previously reported on in the literature is that the location and atmospheric dispersion conditions are far more realistic, e.g., the location could easily resemble a park in an urban area.

Although from the on-set of the project the European Commission classified this research and its outcome, the information presented in this work as well as the set of guidelines derived from these experiments are open to be used by those concerned with the safety and security of first responders while performing their duties in the aftermath of a terrorist attack involving radiological materials.

PS6 (T6.2-0141)

Disaster Management and Control: A Stochastic Process Approach

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Uncertainty is an important characteristic in many real-life settings. Crises and disasters are two of them. They bring about conditions that could lead to undesirable economics or unacceptable HSE state of affairs.

Disaster management stands for processes and procedures aimed at dealing with a disaster in a manner that minimizes damage and allows the affected location or organization to recover quickly. Critical industries, such as nuclear power generation plants, should provide resources and means of improving the efficiency of their disaster response and preparedness.

Effective disaster management requires an effective crisis management and control plan. In disaster settings, the exact times and magnitudes of the events cannot be predicted with certainty. They are probabilistic events, and require corrective intervention measures. Intervention requires the optimal allocation of scarce resources. Therefore, researchers recognize the need to establish a cost-effective (efficient) management and control of disaster events.

In general, published literature in this area seeks to provide more accurate and/or better technological infrastructure of systems to support disaster recovery efforts. However, there is a need to examine key features of stochastic systems of natural events that may affect the decision-making process in nuclear and radiological disaster monitoring, control and recovery, in order to release pressure on operators in NPP control rooms.

This paper treats the inventory of *direct* losses of a disaster as a stochastic system (human casualties, property, destroyed, damaged and recoverable material and equipment). It proposes the Queuing Theory (a Markovian birth-and-death stochastic process) to model the inventory system of disaster events or outcomes. This is a stochastic method, used extensively in Operations Research to model queuing and inventory systems, machine and facility maintenance and repair processes, energy sector, health care systems, inventory management, location of facilities and plants, production scheduling and control, equipment selection and replacement, maintenance and availability, logistics, materials handling & supply chain management, military, defense and security, mining industry, quality control and inspection, sustainability and the Environment, Telecommunications and Information Technology, bank services, and population dynamics.

Here, the Queuing Theory is used as a model for the analysis, management and control of disaster recovery efforts. Service centers are assumed to receive customers arriving and serviced in a Poisson probability distribution. That is, Poisson distribution is used to analyze a system of relief service centers (disaster outcomes) as a queuing system in which both arrivals and service of outcomes are assumed completely random in time, with parameters:

- λ: mean rate of arrival; equal to 1/E[Inter-arrival-Time], where E[.] denotes the expectation operator.
- μ: mean service rate; equal to 1/E[Service-Time].
- $\rho = \lambda/\mu$ for single server queues: utilization of the server; also the probability that the server is busy or the probability that someone is being served.
- c: number of service centers.

The design of a queuing system to manage and control the inventory of disaster outcomes is optimal when a steady-state prevails. If the number c of required service centers is to be determined, the procedure starts with determining the smallest integer c such that the "service center utilization factor"= <1 and to study the resulting values of the corresponding "measures of effectiveness" until a specific measure (such as the "waiting time") is obtained that is acceptable by the Disaster Management Center or Disaster Monitoring Organization from an economic or humanitarian disaster recovery point of view.

The numerical examples given in this work provide insight into the problem that may not be obvious intuitively.

PS6 (T6.2-0401)

A Citizen Science Approach For Dose Rate Mapping In A Contaminated Territory: Dose Rate Results, Analysis Of Participants' Comments And Perspectives

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In the framework of the TERRITORIES European project, a map of the ambient dose rate of a village, the Komarin village, located at the edge of the Chernobyl exclusion zone, was made using the OpenRadiation system (www.openradiation.org) by schoolchildren aged 14-17. During a first meeting with the children, the functioning of the system was explained and children stared to measure the next day. No specific instructions were given, excepted to avoid taking risks and to describe as possible the way the measurements were made. Children appropriate very rapidly the measurement system since 80 measurements appeared in four days on the map of the web site, and 645 measurements were made in one month.

The measured dose rates were not that much different in the Komaryn village as compared to other places in the world, and especially in France. This showed that the radiological situation in the village is safe. However, some hot spots were identified. Discussions with the villagers and students allowed identifying these hotspots as temporary ashes storage. This was the occasion to recall daily life recommendations about the misuse of ashes as fertilizers for kitchen gardens. Thereafter, on the basis of a questionnaire, it was possible to define more precisely the expectations and feeling of participants with the use of the OpenRadiation system. This showed that they wish to share their measurements and to discuss them with other users of the system. Moreover, they indicate that this experiment allowed us to regain control on their environment. This study demonstrates that even 30 years after the Chernobyl accident, the population already have concerns about the radiological quality of their environment, which in turn asks for the methods to be used to maintain the awareness in a post-accidental situation on the long term.

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PS6 (T6.2-0648)

Advanced Atmospheric Dispersion Techniques: Tools at the Forefront of Radiation Protection in Modern Day Dose Assessments or Tools for Creating Aesthetic Aids?

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The perception of risk from nuclear disasters is one of the most prominent barriers to the use of new nuclear technology. In addition, the fear of fallout from accidental discharges is a chief concern. It is the nuclear operator's duty to have adequate response arrangements to avert any possible radiological consequences. Modelling of the dispersion of radionuclides is critical in order to understand the nature of the hazard and identify potential urgent protective actions. The process of predicting plume behavior is complex and further complicated by the requirement for subsequent dose assessment.

This paper provides a comparison of the traditional Gaussian atmospheric modelling approach to modern day alternatives and evaluates the utility of modern day alternatives for radiation protection purposes. A number of state-of-the-art dispersion modelling techniques are considered, including Public Health England's PACE Software[1] and the UK Met Office's NAMEIII[2]. While these models have greater scientific validity, uncertainties in key parameters still dominate the dose estimates. The use of more complex modelling techniques has the potential to lead to the dominant parameters becoming lost in the plethora of variables involved with calculating dispersion and dose uptake. This paper looks at the uncertainties and sensitivities of these codes to their input parameters and the potential impact on the preparedness and response in nuclear emergencies. A review of the usefulness of the visual outputs produced by modern modelling techniques for emergency planning purposes is provided, along with a discussion on the potential impact of these visual aids for public radiation protection.

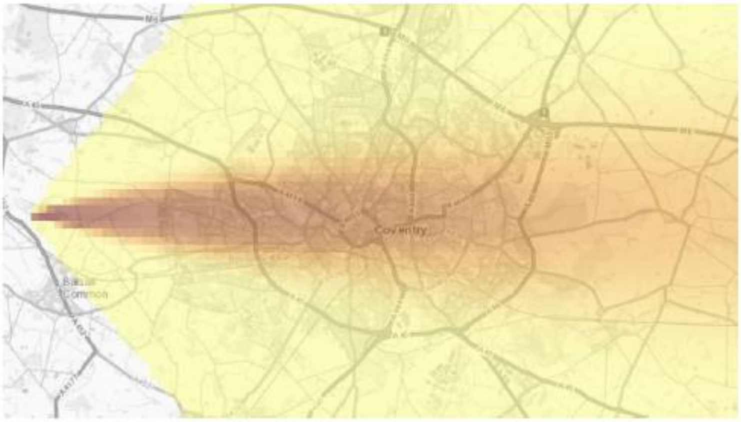

Fig. 1. Dose contour resulting from a fictitious release of ^{60}Co on the outskirts of a large English town.

***Keywords**: atmospheric modelling dose assessment, nuclear emergency*

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PS6 (T6.2-0652)

Radioecology of the Danube River - 50 years Research and Monitoring

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In the 20th century, spacious environmental radioactive contamination due to atmospheric nuclear weapons testing as well as nuclear reactor accidents have had their radioecological impact on the ecosphere and this includes, quite prominently, freshwater resources. Radioecological research in the Danube region began in the nineteen sixties by monitoring ^{90}Sr, ^{137}Cs and other man-made radionuclides. Simultaneously, monitoring of ^{137}Cs and ^{90}Sr in drinking water, soil and agricultural food products (e.g. crop, milk) had been established in all countries of the Danube river basin, as important part of the monitoring program on environmental radioactivity. Very early, also naturally occurring radionuclides such as ^{210}Pb, ^{210}Po, ^{226}Ra, ^{228}Ra, ^{228}Th and ^{238}U originated from industrial plants processing natural materials (e.g. mineral raw materials, building materials) were also included in the environmental monitoring programmes.

The essential objective of the spacious and long-term radioecological monitoring is the full protection of the environment against harmful radioactive exposure in the future to manage sustainable use and conservation of the Danube freshwater resources. Elevated levels of radionuclide concentration in rivers lead to increased health risks for the public drinking processed river water or consuming water animals. The use of contaminated river water for irrigation can also increase health risks by consumption of the agricultural products produced in the irrigated areas. Therefore, it is of importance to monitor continuously the radioecological status of the Danube river ecosphere and to evaluate the impact of artificial and natural radionuclides on the health of the population living in the basin (1).

In this paper, a complete review on radioecological research and radioactivity monitoring carried out in the Danube freshwater ecosystem in the last 50 years is presented. Results of radiometric analysis of Danube water and bottom sediment, collected continuously by sediment traps and additionally by grab sampling during Danube research cruises, are given and discussed. Sample collection techniques, sample preparation and radio-analytic methods, developed and applied in radioecological studies on the Danube river, are shown comprehensively. Additionally, this paper aims to evaluate and visualise the spatial and long-term temporal development of ^{40}K, ^{90}Sr, ^{137}Cs, ^{226}Ra, ^{228}Ra, ^{228}Th, ^{238}U and ^{210}Pb radionuclides in Danube riverbed sediments. Finally, the health risks on the population due to the radioactive contamination of the Danube ecosystem is assessed.

Keywords: Environmental monitoring, radioactivity, public exposure

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PS6 (T6.2-0753)

Environmental Radiation Monitoring in Fukushima i) Large-Scale Survey

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Nine years after the Fukushima Dai-ichi Nuclear Power Station (FDNPS) accident, various environmental radiation monitoring (ERM) are ongoing. In Japan, the Nuclear Regulation Authority (NRA) has a duty of ERM around FDNPS. Airborne radiation monitoring using a manned or unmanned helicopter[1], carborne monitoring and ground-based monitoring[2] have been conducted for dose rate evaluation. These monitoring results have been utilized as basic data for decision-making such as demarcation of evacuation or decontamination areas. We have prepared a series of five presentations regarding ERM in Fukushima. In this article, an overview of the monitoring results is summarized. In addition, lessons learnt through this experience are described. Other presentation topics in the series are on environmental radiation monitoring concerning ii) urban environment, iii) ground, iv) water systems, and v) water bed using special measurement techniques.

Understanding the characteristics of each radiation measurement technique is important for the evaluation of the distribution of the air dose rate. For example, the air dose rate of carborne monitoring have a tendency to be lower than other monitoring results because carborne monitoring measures on the road. On the other hand, airborne monitoring can measure the whole areas including the forest and mountain. However, measurement data of airborne monitoring are relatively rough because of large distances between the detector and the measurement object.

Temporal change of dose rate, expressed as an effective and ecological half-life, is evaluated because the large-scale monitoring results from national projects were accumulated during 9 years after the FDNPS accident. Typically, this tendency of decreasing rate is divided into a fast and a slow component. The value and ratio of these components differ by land-use. The decreasing rate of paved surfaces was faster than non-paved ones based on comparison of airborne and carborne monitoring. Analysis of large-scale monitoring data is important to grasp the influence of the FDNPS accident and to evaluate of radiation exposure of inhabitants.

***Keywords**: Fukushima Daiichi Nuclear Power Station accident, Airborne radiation monitoring, Ground-based radiation monitoring,*

ACKNOWLEDGMENTS

This report summarized the results of the distribution-mapping projects and airborne radiation monitoring projects that JAEA carried out as commissioned business by the Nuclear Regulation Authority.

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PS6 (T6.2-0735)

Environmental Radiation Monitoring in Fukushima ii) Urban Environment

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While most of the area affected by the Fukushima Dai-ichi Nuclear Power Plant (FDNPP) accident is covered by forest, residential areas, which have large impact on public exposure, were also contaminated by radionuclides, especially radiocesium. Current situation and prediction of the radiocesium contamination and air dose rate in the residential areas are a large concern for public and government. Residential area is a multimedia-environment composed of diverse surfaces such as pavements and buildings in addition to permeable fields like agricultural lands. Since radiocesium behavior depends on the surfaces[1], their distribution and time dependency differ greatly from those in other land uses. Therefore, detailed monitoring of radiocesium and air dose rate focusing on the residential areas is necessary to obtain effective and practical information for radiation protection. This presentation shows monitoring tools we employed to residential areas after the FDNPP accident and some findings obtained by the monitoring. For the survey of air dose rate, various tools have been employed like air-borne survey using an unmanned helicopter[2], ground-based surveys (car-borne and backpack surveys) using a γ-ray survey system (KURAMA system)[2], and observation at fixed points using a handy survey meter such as NaI scintillator[3]. These methods were characterized by different monitoring efficiency and spatial resolution, and were employed depending on objects of survey. Air-borne survey can cover extensive area with high monitoring-speed even in the area inaccessible by walk and was useful for distribution mapping. Ground-based surveys need longer time to cover the wide area due to slower velocity of survey, but can provide high spatial resolution. These tools are useful to obtain important information such as areal distribution of air dose rate, which enable to confirm decontamination results and to evaluate time dependence of air dose rate. Especially, faster decrease in air dose rate in residential areas than in other land uses is critical information for the estimation of further exposure. Monitoring of radiocesium activity per unit area (Bq m^{-2}, hereafter inventory) on the surfaces were carried out using a portable Ge γ-ray spectrometer, which is a nondestructive method to detect γ-ray emitted from object surfaces. Artificial surfaces such as pavements and roofs showed smaller inventories than those on permeable fields, and rapid removal of radiocesium from residential areas was also observed even without decontamination. These results, along with the fast decrease in air dose rate in the residential area, suggest a large self-cleaning capacity of urban environment more than the other land uses.

***Keywords**: Fukushima Dai-ichi Nuclear Power Plant accident, Urban, Monitoring*

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PS6 (T6.2-0742)

Environmental Radiation Monitoring in Fukushima iii) Ground

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After the nuclear accident, the information about depth profiles of radionuclides are essential to evaluate the deposition and air dose rate of nuclide. In the case of FDNPS (Fukushima Dai-ichi Nuclear Power Station) accident, radiocesium (^{137}Cs) was major issue for protection of external dose exposure because of its relatively long half-life and physicochemical properties. Immediately after the FDNPS accident, radiocesium was deposited on surface soil. The depth profiles of radiocesium under various land uses have been investigated around FDNPS. In the paddy field, migration of radiocesium from surface to deeper layer of soil with time was reported[1]. However, most studies about the depth profiles of radiocesium focus on the variation of inventory of radiocesium with time over a limited area. There are few reports about the depth profile of radiocesium in soil at wider scale[2]. In this paper, we described the depth profiles of radiocesium in soil around FDNPS to know the reason for the varying distribution of radiocesium.

We have investigated the depth profiles of radiocesium in soil using a scraper plate within 100-km radius of FDNPS during 2011-2016 (9 times). Soils were collected once or twice a year at 85 points. Soils were divided into 5 or 10 mm intervals, respectively. Each layer of soil was placed in a plastic container. The activity of radiocesium was measured by a Ge semiconductor detector. The depth profiles of radiocesium in soil were classified into three types based on following two information; 1): the decreasing trend of radiocesium concentration with depth and 2): the depth (D_p) where the activity concentration of radiocesium is maximum in the layer of soil. In general case, radiocesium concentration decreases exponentially with depth from surface (Type A). In some cases, shift of maximum depth D_p to deeper layers with time was observed (Type B). In complex cases, the depth profile of radiocesium was disturbed by some human activities (ex; decontamination and construction works) (Type C).

In type A and B, the maximum value of D_p is 1.25cm. It is indicated that radiocesium has remained in surface soil where no disturbance has occurred. However, the distribution of type A decreased while that of type C increased over time. The mean value of D_p for all types in each year increased over time. It is estimated that the areas affected by human activities are expanding. The more time passes from the accident, the harder it will be to express the depth profile of radiocesium in soil using a parameter called "relaxation mass depth (β)".

Keywords: *radiocesium, soil, depth profile*

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PS6 (T6.2-0743)

Environmental Radiation Monitoring in Fukushima iv) Water System

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In water systems, most of radiocesium derived from the Fukushima Daiichi Nuclear Power Station (FDNPS) accident exists in particulate and dissolved forms. Especially, understanding the origins and dynamics of dissolved ^{137}Cs is important for clarifying ^{137}Cs transfer from water to crops and aquatic biota because dissolved ^{137}Cs exhibits mobility and high bioavailability. In the present study, we collected and analyzed a suite of time-series water samples from 2014 to 2019, for ^{137}Cs concentration in rivers and a reservoir in Fukushima Prefecture where the catchment has a high ^{137}Cs inventory. This study aims to reveal the trend in long-term variation and the mass balance of dissolved ^{137}Cs for the water system following the FDNPS.

Water samples were collected from the inflow, outflow and in-between the Ogaki Dam Reservoir, located 17 km to the northwest of the FDNPS and situated within the middle stretch of the Ukedo River. The catchment of the Reservoir has the highest inventory of ^{137}Cs among the catchments of Fukushima Prefecture and is located in the heavily contaminated evacuated area. The average ^{137}Cs inventories in the reservoir catchment ranges from 0.2 to 7.2 MBq m^{-2}, and the total inventory of ^{137}Cs in the reservoir catchment is approximately 2.8×10^{14} Bq. At present, ^{137}Cs concentrations in several river fish species of this catchment are above the Japanese regulatory limit [1]. Water samples were collected at the rivers and the reservoir using a bucket and a submersible pump. Water column profiles of temperature and dissolved oxygen (DO) at 0.1 m intervals were measured using a water quality profiler. Samples were brought back to the laboratory and were filtered through membrane filters of pore size 0.45 μm to remove suspended particulates. The activity concentrations of the dissolved and particulate ^{137}Cs were determined by γ-spectrometry using high-purity Ge-detectors.

The trends of dissolved and particulate ^{137}Cs concentrations in water showed a slow decline with time. The effective ecological half-live (T_{eff}) values[2] for the dissolved and particulate ^{137}Cs concentrations in the reservoir outflow were revealed to be significantly longer than in river inputs. In summer season, thermal stratification and hypolimnetic anoxia were established, and the dissolved ^{137}Cs concentration in the reservoir water column was increased towards the bottom layer. In contrast, the vertical distribution of dissolved ^{137}Cs concentration and water temperature in winter season showed relatively constant values with depth, and hypolimnetic anoxia was eliminated almost entirely by turn over. It is assumed that the mobilization was attributed to ion-exchange displacement of ^{137}Cs from sediments by cations such as NH_4^+ released under anaerobic decomposition of organic matter[3]. It is necessary to continue monitoring water systems to evaluate the long-term effect of releasing ^{137}Cs from sediment to water.

Keywords: *river and reservoir, dissolved and particulate ^{137}Cs, effective ecological half-live (T_{eff}),*

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We would like to express our gratitude to the Tohoku Regional Agricultural Administration Office for the permission to undertake our fieldwork.

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PS6 (T6.2-0775)

Environmental Radiation Monitoring in Fukushima v) Waterbed using Special Measurement Techniques

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Conventional radioactivity measurements for aquatic environments involve sediment coring followed by laboratory measurements with stationary gamma spectrometric systems. The process in long, tedious, and costly resulting in limited sampling which affects the visualization of spatial distribution particularly over wide areas. This presents challenges for long-term monitoring over wide areas impacted by anthropogenic radionuclides as is the case for Fukushima, Japan. In-situ measurements are attractive for long-term monitoring and wide area visualization of radioactivity distribution. The Japan Atomic Energy Agency (JAEA) developed a number of in-situ radioactivity measurement technology for aquatic environments including Plastic Scintillation Fibers (PSF) and spectrometers, Lanthanum bromide ($LaBr_3$) and sodium iodine (NaI), that can be mounted on remotely operated vehicle (ROV) in reservoirs or towed by unmanned surface vehicles (USV) at sea[1)]. They have been deployed in ponds, reservoirs and offshore since 2013 to monitor radiocesium concentrations therein. The monitoring results from these systems can aid the understanding of the behavior of the radionuclide in the aquatic environment at spatial scales that may not be practically achievable by traditional sampling, and will be the subject of the presentation.

The PSF detects the position at which radiation originates using the time of flight differences of radiation-induced scintillation from the production point to the photomultipliers located at both ends of the tube. the PSF is designed to sink underwater and can conform to the surface morphology because the detection element has no mechanical structure. Such characteristics are effective for measuring the distribution of radiocesium in bottom sediment. The water-resistant gamma-ray spectrometers are effective for obtaining relatively detailed areal monitoring with discrimination of the type and energy of radionuclides. In estimating vertical distribution, the depth profile of sediment-associated radiocesium was found to correlate with the intensities of scattered and photo peaks[2)]. The accuracy of these systems was checked by comparing with the results of actual sediment sampling.

Though the in-situ techniques may not be as accurate as traditional sampling, they are a useful tool for obtaining an overview picture over wide areas, enabling the identification of points/areas of long-term radiological concern and aiding selection of sampling points for detailed analysis or monitoring[3)]. Future and ongoing work will focus on improving the accuracy of measurements made by these underwater in-situ systems.

***Keywords**: In-situ radioactivity measurements, radiocesium, aquatic environments*

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PS6 (T6.2-0755)

Construction of Emergency Preparedness System in Korea

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The environmental and health effects by the transport and diffusion of pollutants released into the environment due to a nuclear accident must be evaluated rapidly and accurately for the safety of the surrounding population and ecosystem. After the Chernobyl and Fukushima nuclear accidents, the emergency preparedness systems such as NEARC in USA, RODOS in Europe and WSPEED-II in Japan have been developed and constructed in some organizations. In Korea, a radiological accident preparedness system has been developed to predict the atmospheric/marine dispersion of radionuclides and the following dose for a nuclear emergency since 2012. The system constructed in Korea is composed of atmospheric and marine dispersion models, and dose assessment models. Atmospheric dispersion and marine dispersion models can evaluate the behavior of radionuclides released into the air and ocean from the accident. Also, dose assessment model is applied to evaluate the dose for human by ingestion of contaminated foods in land and marine ecosystems. The basic parameters such as real-time meteorological and oceanic circulation data are received from Korea Meteorological Administration and Korea Institute of Ocean Science & Technology. These models are connected in emergency preparedness system and the system constructed using GUI for users. An constructed radiological assessment system can be used to protect humans and the environment against a nuclear accident in neighboring countries(China, Taiwan, Korea and Japan) or anywhere in the world.

PS6 (T6.2-0916)

Specialized Medical Care for Patients Exposed to Radiation as a Result of the Chernobyl Nuclear Power Plant Accident

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About 800 thousand citizens of the Russian Federation were affected as a result of the Chernobyl accident. Almost a quarter of them were clean-up workers of the consequences of the disaster at the Chernobyl nuclear power plant and most of them were exposed to low doses of ionizing radiation. Now, 30 years after the radiation exposure they have up to 12-15 diseases and they needed in specialized inpatient treatment. Data on their morbidity structure and characteristics need to be generalized and are the basis for improving the organization of specialized medical care. To identify the characteristics of morbidity and the main classes of diseases in those suffered from the Chernobyl accident in the remote period 4195 medical records of the Chernobyl clean-up workers were analyzed. All patients underwent inpatient treatment in Nikiforov Russian Center of Emergency and Radiation Medicine, EMERCOM of Russia in 2016–2018. The features of the morbidity of those affected by the Chernobyl accident are founded. The main classes of diseases are presented. The features of the organization of specialized medical care have been identified, in particular, the need has been substantiated for the provision of not only specialized therapeutic, but also specialized, including high-tech, surgical care, as well as medical rehabilitation in the preoperative and postoperative periods, after severe injuries and somatic diseases. The need for the creation and use of unified standards for the provision of specialized therapeutic treatment to those affected by the Chernobyl accident was confirmed. The features of the morbidity rates and the main classes of diseases revealed in the long-term period are the basis for planning the types of specialized medical care for this cohort.

Keywords: *Chernobyl accident; clean-up workers, specialized medical care.*

PS6 (T6.2-0944)

Estimation on Ingestion Dose from Agricultural Products in Case of a Nuclear Accident at a Korean Nuclear Site

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Long-term behaviour of radioactivity in soil, crops, and ingestion doses due to animal products and vegetable were estimated assuming that 12 PBq of Cs-137 released into the atmosphere from a Korean nuclear power plant. The air transport of radioactivity was estimated by the AERMOD, an atmospheric dispersion model developed by the US EPA. The human ingestion dose due to crop contamination and livestock products was estimated by modifying the PATHWAY [1]. Fig. 1 shows the Cs-137 deposited during the accident. It can be seen that Cs-137 deposition appears high in the east due to the west wind dominant. Areas with radioactivity concentration in soil in excess of 5,000 Bq/kg (equivalent to 2.46E+06 Bq/m^2) were found to be 8 km eastward and 11 km northeast from the power plant site. Cs-137 deposited on the vegetable surface and soil surface at the time of the accident moves into labile soil and into vegetable internal. Approximately three years after the accident, the proportion of radioactivity in each compartment reaches an equilibrium state. The radiation from the intake of meat was the highest, followed by vegetables, milk and respiration. It was shown that ingestion dose was high in the first year after the accident. There was no significant change from then on. This is believed to be due to the fact that radioactive inventory of labile soil has not changed significantly since the first year of the accident. Therefore, it is deemed possible to propose countermeasures to prohibit the consumption of agricultural and livestock products for one to two years after the accident, or to selectively prohibit the consumption of meat, milk and vegetables by taking into account the radiation exposure caused by the eating habit.

Fig. 1. Deposition density of Cs-137

Keywords: *Ingestion dose, radiological accident, countermeasures*

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PS6 (T6.2-1001)

Preliminary Study on Interference Cancellation of Indirect Gamma Spectroscopy Method for Responding Nuclear Emergency

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Nuclides that emit alpha particles can occur a very large internal exposure risk in the event of a nuclear emergency. Therefore, fast analysis of nuclide such as uranium, thorium, and plutonium, which emit alpha particle could be necessary. (Especially when nuclear emergency occurs) In order to precisely analyze the nuclide emitting alpha particles, it is necessary to apply the alpha spectroscopy technique. However, there are some disadvantages to performing alpha spectroscopy, such as requiring of highly skilled analysts, a lot of labor for pretreatment, long analysis time, and high cost of analysis. Therefore, there are limitations in using alpha spectroscopy in emergency situations. The gamma spectroscopy method, on the other hand, does not require complex pretreatment of the sample. For this reason, repeated measurement of a sample is also possible. And it doesn't take long to get the results. In addition, the gamma spectroscopy method can be performed at a much lower cost than the alpha spectroscopy method. Therefore, in order to overcome the shortcomings of the alpha spectroscopy method, we conducted a preliminary study that summarizes the theory that can rapidly analyze uranium, thorium, and plutonium nuclide by indirect gamma spectroscopy method. U-235 emits 185.7 keV of gamma while simultaneously emitting alpha particles. However, it has an interference relationship with 186.2 keV of Ra-226 which coexists in the decay-chain of U-235. Therefore, in order to obtain accurate measurement results using the 185.7 keV of U-235, it is necessary to calculate the net count of 185-186 keV region of Ra-226. To calculate this net count, we need to utilize 295.2 keV of Pb-214 in equilibrium relationship with Ra-226. This is expressed as equation 1-3. **Equation (1):** Activity (Pb-214) = Count (295.2 keV) / [Efficiency (295.2 keV) * Yield (Pb-214, 295.2 keV)] **Equation (2):** Count (Ra-226, 186.2 keV) = Activity (Pb-214) * Efficiency (186.2 keV) * Yield (Ra-226, 186.2 keV) **Equation (3):** Activity (U-235) = [Total Count (185 - 186 keV) - Count (Ra-226, 186.2 keV)] / [Efficiency (185.7 keV) * Yield (185.7 keV)]. Th-232 and U-238 also emit alpha particles, while simultaneously emitting gamma rays in the 63keV region. Therefore, in order to obtain accurate measurement results of Th-232 and U-238, interference cancellation technique and equilibrium theory in 63 keV region is required. This is expressed in equation 4-7. **Equation (4):** Activity (Th-232) = Activity (Ac-228) by equilibrium theory = Count (911 keV) / [Efficiency (911 keV) * Yield (Ac-228, 911 kev)] **Equation (5):** Count (Th-232, 63.9 keV) = Activity (Th-232) * Efficiency (63.9 keV) * Yield (Th-232 63.9 keV) **Equation (6):** Count (Th-234, 63.39 keV) = Total Count (63 keV) - Count (Th-232, 63.9 keV) **Equation (7):** Activity (U-238) = Activity (Th-234) by equilibrium Theory = Count (Th-234, 63.39 keV) / [Efficiency (63.39 keV) * Yield (Th-234, 63.39 keV)]. In addition, Pu-239, which emits alpha particles, also emits 204.06 keV gamma rays, which can be analyzed using the indirect gamma spectroscopy method. However, having an interference relationship with 203.55 keV of Am-241, it would be necessary to utilize interference cancellation techniques and equilibrium theory, such as the equations applied to U-235 and Th-232. To follow up the theoretical considerations of this study, we present a comparative study by results of alpha spectroscopy and indirect gamma spectroscopy. As a result of the follow-up study, the reliability of the indirect gamma spectroscopy analysis can be verified. Ultimately, indirect gamma spectroscopy can be used to quickly obtain analytical results, which will enable the use of alpha-emitting nuclide concentration in OIL for nuclear emergency.

ACKNOWLEDGMENTS

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PS6 (T6.2-1027)

The Development of Emergency Condition Assessment System for NPP

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For meeting the requirements of preparedness and response in case of a nuclear emergency in nuclear power plant(NPP), a system, which used to assess the reactor emergency condition, such as, assess core damage status and source term released to the environment, must be developed. This paper introduced the development of emergency condition assessment system for NPP, especially for pressurized water reactor, including its software infrastructure, assessment modules, platform and data interfaces, etc.The system adopt the JAVA language as the platform, and open source MySQL as the data management system, supporting Windows and Linux OS. This paper detailed introduce the related subsystems, 1)basic data acquisition subsystem, 2) core damage assessment subsystem, 3)source term estimate subsystem, 4) assessment results display subsystem, 5)user authorization management subsystem. The system can real-time acquiring the reactor operation parameters from Distributed Control System(DCS), assess the core damage status and calculate the amount of radioactive material released from the core to the primary coolant system, subsequent to the containment and environment..

This paper introduces the development of reactor emergency condition assessment system for NPP(ECAS-NPP), including the overall design, platform development and interface design, module development, etc. The system can diagnose the emergency condition of the reactor core based on the operation data of facilities, give the qualitative and quantitative results of the core damage, and the amount of source term released under different release paths.

The main functional subsystems of the system include: basic data acquisition subsystem, core damage assessment subsystem, released source term estimation subsystem, result display subsystem and personnel authority management subsystem. The core damage assessment is mainly based on the emergency condition parameters provided by the basic data acquisition subsystem to evaluate the core damage state and degree, and the result provides input for the released source term estimation subsystem.The main business processes are shown in Figure1.

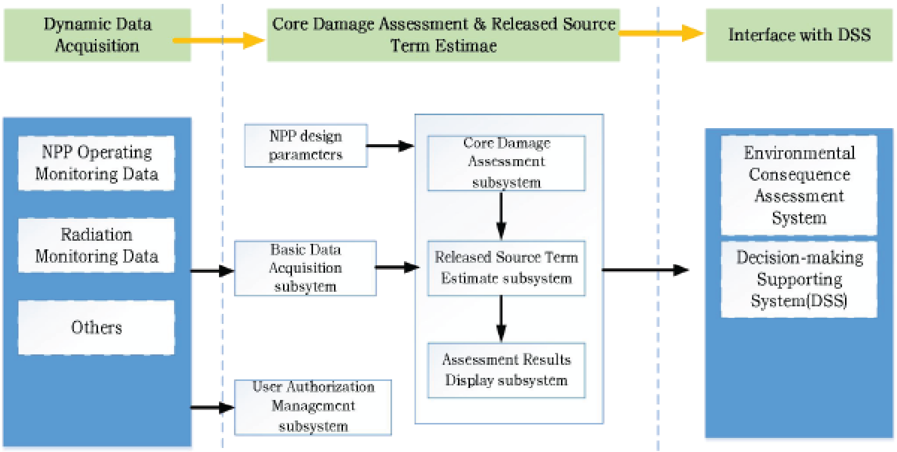

Fig. 1. Business process flow of ECAS-NPP.

The output of the system includes two categories: core damage assessment results and released source term estimation results. The core damage assessment results include qualitative and quantitative results, qualitative including core damage state, such as no damage, fuel damage, core melting, etc., quantitative results for the extent of core damage. The released source term estimation result includes the core total source, the core

PS6 (T6.2-1027)

The Development of Emergency Condition Assessment System for NPP

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damage portion, the potential release source term, the source term released to the primary circuit, the source term released to the containment, The source term released to the environment by the containment, the source term released to the environment by the steam generator tube rupture(SGTR), the total source term released to the environment.

Fig 2. Interface of core damage assessment based on real-time online monitoring instrument readings.

Keywords: *core damage assessment; source term estimate; nuclear emergency; HPR1000*

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PS6 (T6.2-1089)

Multi-Model Ensemble Dispersion Prediction System for Nuclear Emergency Response

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The reliability of dose assessment system depends on the accuracy of the weather and dispersion prediction models. There are several methods to increase the accuracy of these models, including improvement of initial conditions, dynamics, and physics processes, and application of multi-model ensemble prediction technique. The most effective and reliable way to improve the accuracy of the models is to utilize multi-model ensemble prediction technique, in spite of disadvantage that require high-performance computers.
To provide credible and reliable dispersion prediction information in case of nuclear accident in neighboring countries, the multi-model ensemble dispersion prediction technique is applied to the Accident Dose Assessment Modeling system (ADAMO) which was developed for a nuclear emergency response in Korea. Using three global meteorological forecasting models (KMA-UM, NOAA-GFS and CMC-GDPS) and two atmospheric dispersion models (FLEXPART and HYSPLIT), six meteorology-dispersion combinations are constructed. The multi-model ensemble mean outperforms most of the individual models with smaller biases and higher correlation on the spatial and temporal variation of wind fields. If methods such as performance-based ensemble mean and selected ensemble members are considered, the accuracy of the prediction can be further improved. In addition, uncertainty of the dispersion prediction and various prediction scenarios can be provided to decision makers, ultimately increasing the reliability of the system.

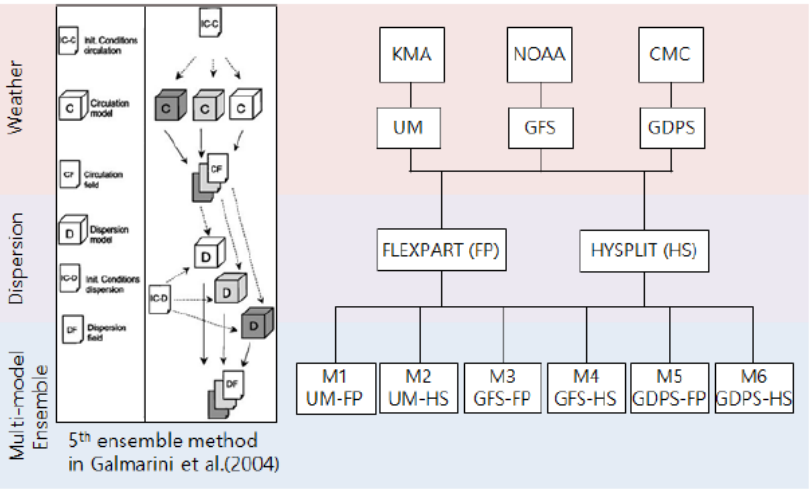

Fig. 1. Composition strategy of multi-model ensemble dispersion prediction system

***Keywords**: Nuclear emergency response, Multi-model ensemble, Dispersion prediction*

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PS6 (T6.2-1090)

Risk Perception of Radiation Exposure and Compared with Their Risk Perception in Tomioka Town and Kawauchi Village

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After the TEPCO's Fukushima Daiichi Nuclear Power Station (FDNPS) on March 11, 2011, almost all residents of Tomioka Town were forced to evacuate. On April 1, 2017, six years after the accident, the Japanese Government lifted the evacuation order of Tomioka, excluding the "difficult-to-return zone". However, the number of residents who have returned to their homes is limited. As of October 1, 2018, only 791 (6.0%) of the 13,132 registered residents have returned to Tomioka town [1]. Previously, we investigated the ITR of residents in Kawauchi village, which is located within 30 km of the FDNPS and was lifted evacuation order in March 2012 (one year after the accident). As of April 2018, 2,197 of 2,713 (80.9%) residents had already returned. In this study, we compared the risk perception of radiation exposure in residents of Tomioka town and Kawauchi village. In 2017, 46.9% residents of Kawauchi village answered that health effects would occur in children and 30.9% answered that genetic effects would occur. On the other hand, 77.6% residents of Tomioka answered that health effects would occur in children by living in Tomioka, and 71.8% answered that genetic effects would occur by living in Tomioka. These results suggested that many residents of Tomioka have anxieties about the health effects of children and genetic effects by residing in Tomioka. For the recovery of the community in Fukushima, risk communication about radiation exposure and the health effects considering the background of each resident is essential.

***Keywords**: Fukushima Daiichi Nuclear Power Station; intention to return; radiation; risk perception; Tomioka town*

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PS6 (T6.2-1168)

Dose Reduction Factor of Decontamination for the Use in Dose Assessment after the Fukushima Daiichi Nuclear Power Plant Accident

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Decontamination is one of protective actions for reducing radiation doses to the public living in the areas contaminated by radioactive materials. In general, effectiveness of decontamination is represented by decontamination factor (DF) evaluated from the reduction of surface density or surface radiation dose rate. However, the DF cannot be used directly for human dose assessment because those represent the reduction of doses and radioactivity for decontaminated surface only. If the dose assessment were performed taking into account the effects of decontamination, other type of dose reduction factor (DRF) which is evaluated from dose rate in a room or dose equivalent to human body is needed.

In this study, we evaluated the DRFs of decontamination for dose assessment based on the experiences of the Fukushima Daiichi Nuclear Power Plant (FDNPP) accident. The DRFs were evaluated as the ratio of the ambient dose equivalent rate in a room after decontamination to before that. The ambient dose equivalent rate was calculated using Monte Carlo particle transport simulation code PHITS taking into account the contribution from roof, outer-wall and the ground. The contributions from the surfaces were calculated using a model of typical Japanese wooden house (Furuta and Takahashi, 2014) on the assumption that the surface was contaminated by ^{137}Cs and ^{134}Cs with 1 Bq/m^2, respectively. The radioactive cesium were distributed uniformly on the ground within a radius of 500 m from the house model. The house model was located in the center of a half-sphere which is filled with air. For roof and wall, relative surface intensity of 0.85 and 0.01 were used assuming wet deposition condition (Jones et al., 2009; Yoshimura, 2014).

Results of our calculations are shown in **Table 1**. From this table, the DRF is evaluated as $\sum_i c_{i,1}/\sum_i c_{i,0}$, where, index, i, means type of surface. Indices of 0 and 1 mean before and after decontamination, respectively. The parameter of $c_{i,1}$ is calculated by $c_{i,1} = DF_i \cdot c_{i,0}$. The DFs is obtained from the debrief report of decontamination work and a previous study after the FDNPP accident (JAEA, 2012). As the results, the DRF for decontamination after the FDNPP accident was evaluated as 0.76. This is the averaged value for the first floor and second floor of the model of typical Japanese wooden house.

Table 1. Decontamination factor and contribution to dose rate inside of a house

Surface, i	DF_i (1),(2)	Contribution to dose rate inside of a house, $c_{i,j}$, (μSv/h)/(Bq/m^2)	
		Before decontamination	After decontamination
Roof	0.88	1.90×10^{-6}	1.67×10^{-6}
Wall	0.60	3.91×10^{-8}	2.34×10^{-8}
Ground	0.18	6.33×10^{-6}	4.62×10^{-6} (3)

(1) JAEA (2012). (2) Debrief report of decontamination in Namie town in 2015. (3) This value was calculated based on the assumption that decontamination are performed for the ground in a 10 m square area around a house.

Keywords: Fukushima Daiichi Nuclear Power Plant accident, decontamination, effectiveness

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PS6 (T6.2-1187)

Environmental Radiation Monitoring in Fukushima vi) External Dose Evaluation Based on Monitored Air Dose Rate

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Nine years after the Fukushima Dai-ichi Nuclear Power Plant (FDNPP) accident, external exposure from radionuclides deposited on the ground is the main exposure pathway[1]. There are some methods to evaluate external exposure dose. Just after the accident, the Japanese government had been applied a simple model (supposing spending 8 hours outdoors and 16 hours indoors, using dose reduction factor (0.4) of indoor ambient air dose rate, and estimation based on integrated ambient dose equivalent). This simple model can evaluate public exposure dose safely for the decision making, however it was not sufficient to evaluate individual exposure. In 2013, the Nuclear Regulation Authority (NRA) showed that individual exposure dose is important as practical value to control exposure when residents return into the zone of which evacuation order is lifted. Using personal dosimeter is common method for measuring individual exposure dose however there are some difficulties. For example, personal dosimeter measurement can't be applied in the area restricted to enter, or it's a burden to wear personal dosimeter always. Therefore, semi-theoretical models, which can evaluate individual exposure easily based on measured air dose rate, have been developed. This study shows these methods to evaluate external exposure dose (i.e. governmental simple model, personal dosimeter, and semi-theoretical model) and quantitative relationship among those methods.

The semi-theoretical model evaluates external dose based on the ambient air dose rate map and the length of time spent at each location by installing dedicated software to the smartphone. Dose reduction factor 0.4 and age-dependent conversion factors from ambient dose equivalent to effective dose are also used. To clarify quantitative relationships among the methods, we collected individual exposure dose and activity pattern data (location corresponding with time) of people who lived or worked in the area near FDNPP using personal dosimeters (D-Shuttle; Chiyoda Technol Corporation, Tokyo) and smartphones, respectively, in 2018 and 2019. Ambient air dose rate used in the model was obtained from the ambient air dose rate map, which is created by multiscale Bayesian data integration approach[2] based on multiple survey data.

In comparison between external doses of each method, the results of the governmental simple model tend to overestimate those of personal dosimeter, and the results of the semi-theoretical model and those of personal dosimeter have a correlation with some amount of errors. Though personal dosimeter can measure personal doses directly, simple model and semi-theoretical model need assumption using some parameters, so those parameters contribute to accuracy and errors. There are also uncertainties about values of personal dosimeters especially when exposure doses are low. These results contribute to lift remaining evacuation zone in future.

***Keywords**: Fukushima Dai-ichi Nuclear Power Plant accident, External exposure, Effective dose*

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This report used the results of a project that JAEA carried out as commissioned business by the Nuclear Regulation Authority.

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PS6 (T6.2-1187)

Mass Interception Factor and Weathering Half-lives of Iodine -131 and Radiocaesium in Leafy Vegetables Affected by the Fukushima Daiichi Nuclear Power Plant Accident

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When agricultural crops were directly contaminated with radioactive fallout due to a nuclear accident, it may possible to estimate the activity concentration in edible part of crops using interception factor together with weathering half-life (T_w) at the time of crop sampling. The Fukushima Daiichi Nuclear Power Plant (FDNPP) accident occurred in March 2011 made us possible to provide these factors. To describe the initial radionuclide activity concentration of the vegetables, interception factor, which is defined as the ratio between the amounts of deposited radionuclides retained by the edible parts of the plant (Bq/kg wet mass) and that of total deposited in the area (Bq/m^2) [1]. Because radioactivity monitoring in fresh vegetables were not carried out immediately after the accident started, the initial radionuclide activity concentration of fresh leafy vegetables can be back calculated from continuous monitoring data. For that purpose, T_w values can be used; however, the data for leafy vegetables observed under the field conditions were not reported, thus we calculated T_w values for ^{131}I and radiocaesium using wild herbaceous plants and also using vegetable data from food monitoring conducted by the Ministry of Health, Labour and Welfare, Japan observed in March to April 2011.

To obtain T_w, we firstly calculated effective half-life (T_{eff}) using the following equation: T_{eff}=ln(2)/λ. The T_{eff} is expressed as follows. $1/T_{eff}=1/T_w+1/T_p$, where T_p is the physical half-life of the radionuclide. Using these equations, T_w is then calculated as the following equation: $T_w=T_{eff}\times T_p/(T_p-T_{eff})$. After data collation and calculation, T_w values in leafy vegetables obtained under open field conditions ranged from 3.0-16 d with the geometric mean of 8.4 d, and that for radiocaesium ranged 3.2-10.6 d with an average of 7.8 d. There was no significant difference between T_w values of these radionuclides. Details of the data analysis are provided our recent publication [2]. Then, using the T_{eff} (or T_w) values obtained as above, the initial activity concentrations of ^{131}I and radiocaesium were calculated. We also calculated averaged deposition concentrations at the leafy vegetables collection sites in Bq/m^2 and thus we could calculated the mass interception factors for leafy vegetables. The geometric mean values of mass interception factors of ^{131}I and radiocaesium were 0.072 and 0.099 m2/kg wet mass. These values are useful to estimate the radiation dose by ingestion of leafy vegetables harvested just after a nuclear accident.

Keywords: *Mass interception, Weathering half-life, Leafy vegetables*

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PS6 (T6.3-0288)

Development of a Radiological Safety and Security Risk Index

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Given the rising threat of radiological and nuclear terrorism, it is imperative to assess if radiological facilities, such as universities and medical centers, have the means to fully understand and evaluate the combined safety and security of their radioactive sources. In this context, risk assessment is a function of threat, vulnerability and consequences. This study aims to develop and demonstrate a methodology to compute a risk index for radiological facilities, based on the probability of occurrence of a Threat Event (TE) and its subsequent magnitude of incurred loss. This risk index provides a quantitative value for comparing risk and making decisions towards radiological safety and security improvements. The index employs inputs that include a set of threats, vulnerabilities, and consequences. These were used to construct a single composite number by weighing the threat scenario probabilities, relative attractiveness and characteristics of the radioactive material, multiple parameters elevating vulnerability of source security, and the consequence net loss. The risk decomposition is based on the Factor Analysis of Information Risk (FAIR) ontology. Probability density functions and event trees were then used to simulate scenarios to estimate the probability of successfully completing a malicious act at the university, such as theft of the source. To demonstrate this index, a higher education institution that uses a number of radioactive materials for research and teaching, was analyzed. Specifically, three facilities housing nuclear or radioactive sources at the university were compared: a research reactor facility, Co-60 irradiator, and radiopharmaceutical laboratory. The emphasis of the study is on the research reactor, but the other facilities were also analyzed for comparison. The research reactor facility houses a 10-kW swimming pool type reactor containing plate type uranium/aluminum fuel. The facility also houses other fuel and radioactive sources needed for operations and research. The irradiator facility contains both Co-60 and Cs-137 sources with TBq amounts of activity. The radiopharmaceutical facility contains a number sealed and unsealed sources with GBq amounts of activity. Two proposed safety (equipment malfunction and human error accidents) and security (malicious attack of theft and sabotage) scenarios were simulated for each facility. The radiopharmaceutical laboratory sources yielded the highest probability of both successful sabotage and theft outcomes as well as probability of accident. The reactor facility yielded the highest consequences in the sabotage scenario. The contribution of the proposed research is significant as it allows for a new tool in the field of coupled radiological source safety and security-one that is expected to introduce, analyze and numerically test a methodology that yields a facility level risk index.

PS6 (T6.3-0557)

Use of Monte Carlo Simulation for Efficiency Calibration in Emergency Response using "in situ" Gamma-ray Spectrometry

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This study shows the use of the Monte Carlo simulation Code DETEFF for efficiency calibration in "in situ" gamma-ray spectrometry applied to emergency response. It was studied the influence of detector characteristics in the accuracy of the calibration factor and the obtained values of activities deposited in the field for several artificial radionuclides released in a nuclear accident.

PS6 (T6.3-0567)

Multiple Elements Extraction in Vitro by Cevidra® Cleansing Emulsion Developed for Skin Radiological Decontamination

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Due to the lack of specific and efficient treatment for skin decontamination in case of accidental exposure to radionuclides, a cleansing oil-in-water emulsion was developed in collaboration with a pharmaceutical laboratory and registered as a medical device named Cevidra®. Originally designed for selective chelation of transuranium elements, the active ingredient of the formulation, namely the carboxylic calix[6]arene molecule, was evaluated in the present study for its ability to extract activation or fission products from a contaminated solution. This in vitro chelation study consisted in mixing the cleansing emulsion (7.5 g/L carboxylic calix[6]arene) with an equivalent volume of a multi-elemental solution of a selection of metal cations of interest (manganese, cobalt, strontium, zirconium, silver, antimony, cesium, uranium and plutonium) at concentrations comprised between 1.26 to 1.58 micromol/L in tubes for 5 minutes. Physico-chemical characterizations of the formulation showed that calixarene molecules are distributed and entrap metal cations at the oil-water interface in the emulsion. So, after separation of the phases of the mixture by ultracentrifugation and ultrafiltration, the amounts of the elements complexed with calixarene in the oily phase and free elements in the aqueous phase were determined by ICP-MS spectrometry. The results of ten assays showed a significant effect of calixarene, expressed as a percentage of complexed elements compared to the initial amounts, with high specific extraction yields ranging from 71.2±1.8% for plutonium up to 97.4±0.3% for strontium. Since the mode of action of Cevidra® is different and hence, cannot be compared in the frame of the present study with reference treatments such as DTPA solution or conventional soapy water, further comparison tests should be performed with other classical cleansing creams. However, the excellent results obtained with Cevidra® allowed widening its indications for the emergency decontamination of miscellaneous elements involved in nuclear industry or after possible malevolence act.

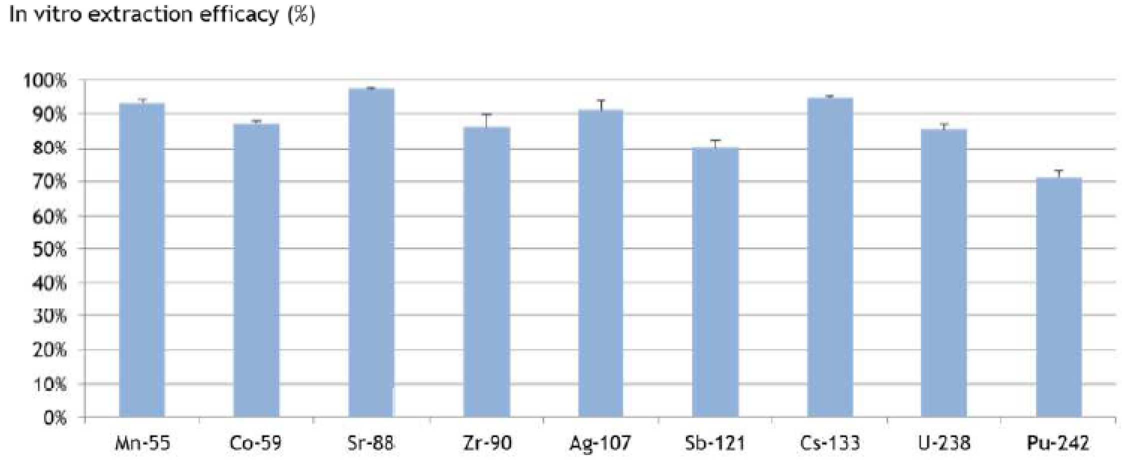

Fig. 1. In vitro extraction of different elements by Cevidra® cleansing emulsion

Keywords: *radionuclides, in vitro extraction, calixarene cleansing emulsion*

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PS6 (T6.3-0636)

Radiation Protection Preparedness using the Decision Support System LASAIR in Case of Malevolent Use of Brachytherapy Ir-192 Sources

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The risk for people dying from radiological terrorism that consists of detonating a nuclear device, or as an attack on a nuclear power plant, is considered to be very unlikely. The risk for exposure from dispersion of radioactive material with conventional explosives is however more likely. The common usage of Ir-192 sources in both brachytherapy and in industry, together with the physical properties of this isotope put high demands on logistics in shipment and handling. There is nowadays a fear of terrorist actions involving dirty bombs. In this context, the Swedish Radiation Protection Authority recently issued a call for improving dosimetric preparedness and competence to cope with large emergencies involving various sources of radioactive material. The aim of this study was to simulate a radiological dispersion device (RDD) using brachytherapy Ir-192 sources as in a malevolent attack in order to increase awareness of the effects of improper disposal, transportation or storage at the clinic.

A scenario in which Ir-192 sources brachytherapy sources temporary stored at the hospital before being picked-up by the vendor for being disposed were stolen was assumed. Two scenarios were simulated: a RDD consisting of explosives as primary mean for the spread of the isotope and the unintentional radiological deployment in case of a car accident followed by fire. The dispersion of the radioactive debris was simulated using the LASAIR system (Lagrange-Simulation of the dispersion and Inhalation of Radionuclides). The selected places for the fire and explosion where decided based on the likelihood to have a high impact on infrastructure, communication and healthcare. The total activity at the date of interest from all sources in storage was calculated and summed up as source input in LASAIR. The source activity is however classified; hence the results are only presented as dose rates and total deposition. The Swedish Meteorological and Hydrological Institute, SMHI, provided with meteorological conditions for the date of interest.

The area where at least 20 μSv effective dose is received in the first two hours have been estimated. If the meteorological conditions are unchanged during an hour after the events, the two scenarios of fire and explosion will result in similar radiological dispersion. The simulations show that a large area of Stockholm will be affected by a RDD of this kind, making the communication of the risks and countermeasures to the public to be of highest priority to minimize short and long term consequences. In case of explosion, as plausible in the RDD scenario, the radioactive fallout was deposited in a shorter time than in the situation of a fire. Wind speed and direction had a large impact on the results. In presence of rain fallout was more concentrated to the coordinates of the explosion or fire. The urban landscape also influenced the deposition of the isotope, although on a local level.

The dispersion of radioactive material from brachytherapy sources simulated with LASAIR results in a relatively large cloud, with low radioactive concentration. Radiological preparedness involving brachytherapy sources can be developed through studying close to real life scenarios simulated in LASAIR.

Keywords: *Malevolent attacks, dirty bomb, brachytherapy.*

PS6 (T6.3-0701)

Radiation Protection Optimization Software for Arctic Nuclear Legacy Site

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This presentation describes the stages of integration of Information-Analytical System (IAS) specially designed to increase the radiation safety of workers during remediation work at the nuclear legacy site in Andreeva Bay in northwestern Russia.
The planning and optimization process includes education and training of personnel, estimation of radiation doses for the upcoming work, preparations for unplanned situations, and implementation of practical safety measures within the targeted radiation-hazardous works.
The IAS allows not only to collect, store and display data from site radiation monitoring systems, but also to solve a number of analytical issues, for example: Construction of radiation situation maps; Optimization of doses during technological operations; Search for optimal evacuation and working routes (based on graph theory algorithms); Transition directly from the ADER to radionuclide surface contamination density at the industrial site or in the premises (the method is based on the numerical solution of the Fredholm equation of the first kind).
The presentation describes radiation situation dynamics at the Andreeva Bay nuclear legacy site over 2002–2016, the period of preparation for the most critical phase of remedial work: removal of spent fuel assemblies from the dry storage unit.
In 2016, in the Andreeva Bay site, Northwest Center SevRAO – Branch of FSUE “RosRAO”; a research emergency response exercise (REE) was conducted during which the effectiveness of the actions of the participants was studied using modern computer methods for modeling the radiation situation. For the first time in an emergency, EasyRAD computer code developed by SRC-FMBC experts is used. This marked the beginning of the exercises series “DOCKING”.
In 2018, in the Andreeva Bay site, an emergency response exercise with international participation “Arrangement and Conducting Emergency Exercise on Interaction of Management Bodies and Attracted Forces of the State Atomic Energy Corporation “Rosatom” and FMBA of Russia in a case of Radiological Accident during Spent Nuclear Fuel Management” took place ("Docking-2018").
The stage of testing the interaction between the operator and the regulator is shown in the development of urgent decisions and recommendations for protective actions for personnel in terms of determining radiation reconnaissance routes, taking into account the minimization of staff doses.
The developed computer system is used to support the optimization of remediation work, as well as regulatory supervision of radiation safety of personnel and is an important element of the emergency preparedness system of the FMBA of Russia in difficult Arctic conditions.

Keywords: nuclear legacy site, Radiation Protection, EPR

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PS6 (T6.3-0701)

Radiation Dose Assessment for the Environmental Release of the Fukushima Treated Water

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As a result of the Fukushima nuclear accident, the contaminated water is currently generated when groundwater and rainwater contact with damaged reactors and fuel debris in buildings. At present, the contaminant water is treated at multiple purification facilities including the Advanced Liquid Processing System (ALPS), and subsequently, the treated water is being stored in tanks at the site of the Fukushima Daiichi nuclear power station, Japan (1, 2). The total amount of treated water being stored is about 1.2E6m^3 as of November 21, 2019. A number of technical options for handling the treated water including the geosphere injection, the discharge to the sea, the release of vapor or hydrogen into the atmosphere, and the underground burial, are currently being discussed (3). Among the options, the discharge to sea and the vapor releases seem to be most promising because those are the experienced technologies.

To investigate the radiological effect for the case of the marine discharge and the vapor release into the atmosphere of the Fukushima treated water, the radiation dose for the public was estimated by applying the UNSCEAR methodology for estimating the public exposures due to radioactive discharge (4). The eight key radionuclides ^{3}H, ^{60}Co, ^{125}Sb, ^{90}Sr, ^{106}Ru, ^{129}I, ^{134}Cs, and ^{137}Cs were considered in the estimation. The mean measured activity concentration of the treated water being stored in the tank was used as the source term of the radionuclides (5). For both handling options, all radionuclides in tanks were assumed to be continuously released into sea or vapor in a year. The considered exposure pathways for marine discharge were internal exposure through the ingestion of seafood and the external exposure from marine sediments. The considered exposure pathways for vapor release into the atmosphere were the cloud immersion (external exposure and inhalation), the external exposure from deposits onto the ground, and the ingestion of terrestrial foodstuffs.

The public exposure dose at 5km from the release point was estimated to be 0.00023mSv and 0.00136mSv for the marine discharge and vapor release into the atmosphere, respectively, which corresponds to 0.023% and 0.14% of the annual dose limit for the public. For the marine discharge, the external exposure from sediments was the dominant pathway, and ^{60}Co was the main contributor of dose, while for the vapor release the internal exposure due to inhalation and ingestion was the dominant pathway, and ^{3}H was the main contributor of dose. On the other hand, for the vapor release into the atmosphere the public dose at the Busan area in the Republic of Korea at about 1090km distances from Fukushima turned out to be so small as negligible (about 0.00000456mSv). The present results indicate that the environmental release of the Fukushima treated water seems to be safe scientifically. However, despite the scientific safety, the continuing political and social disputes for the handling of the Fukushima treated water suggests that the issue is still one of the problems to be resolved in both countries of Japan and the Republic of Korea for public acceptance.

***Keywords**: Fukushima treated water, marine discharge, vapor release, public dose*

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PS6 (T6.3-0803)

Establishment of Reference Levels for Emergency Exposure Situations for an Assuming Nuclear Power Plant Accident Scenario

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In emergency exposure situation, reference level is a prospective and source-based radiation dose criterion for individual from a single source. The reference level is the upper limit of the expected dose in the optimization of protection for the source. In order to set appropriate reference levels, radiation dose assessment for a various accident scenarios should be conducted. In addition, accident consequence analysis studies are required to select an accident dose evaluation point from an optimization point of view. As a beginning step for the establishment of reference levels for emergency exposure situations, radiation doses at offsite were evaluated for an assuming nuclear power plant accident scenario. For this evaluation, we assumed the interfacing systems loss-of-coolant accident (ISLOCA) incident scenario of the Surry nuclear power plant. The scenario was used for PSA as a part of the SOARCA project. Meteorological data of Surry nuclear site in 1988 were used. Wind speeds were divided into 9 categories. Atmospheric stability classes were divided into 6 categories. Using the WinMACCS computer code, off-site dose at 560 m corresponding to domestic NPP's exclusion area boundary (EAB) was evaluated. RASCAL computer code was used to verify the results of WinMACCS computer code. The probabilistic consequence of ISLOCA accidents in Surry nuclear power plant was evaluated using WinMACCS. Figure 1 shows generalized off-site dose according to distance from the accident site using WinMACCS and RASCAL computer codes. The results at EAB by RASCAL computer code were about 1.3 time higher than the results by WinMACCS. The results of WinMACCS and RASCAL at 8 km were about 0.08 and 0.07 times compared with the results at the reference point. The WinMACCS computer code can simulate the phased release of plume caused by an accident. However, the RASCAL computer code can simulate only the single plume. Therefore, the trend of results by distance is similar. However, the difference was about 50% at 3 km. In this study, the deterministic and probabilistic methods were used to evaluate the off-site doses in the event of a nuclear power plant accident. The results could be used as a basis for establishing reference levels in future emergency exposure situations.

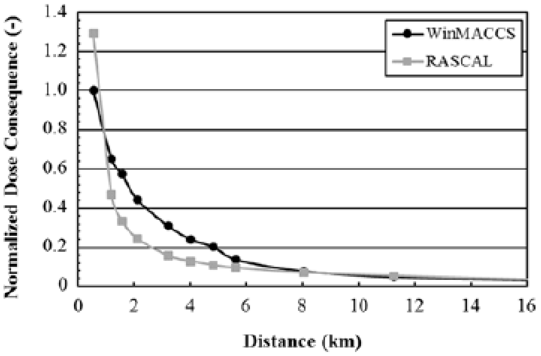

Fig 1. generalized off-site dose according to distance from the accident site using WinMACCS and RASCAL computer code

Keywords: *Reference level, Emergency exposure, Off-site dose, Accident consequence*

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PS6 (T6.3-0828)

Developing a Remote Mapping Device for Radiation Monitoring and Investigation

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Over the last few years, the demand for unmanned system has been raised for rapid and safe response to some nuclear disaster like the Fukushima accident or the dispersal of radioactive materials caused by unauthorized acts. Typically, a gamma or neutron detector mounted on a drone is used for radiation surveillance but it suffers from lower detection efficiency related to short operation time of a drone and allowable altitude. In order to acquire the proper data, it is required to place a detector as close to the source term as possible and maximize the detection time. Therefore, the present authors have designed a radiation mapping device named the gamma probe consisting of a CsI sensor, electronics for wireless data transmission and software. A parachute is also designed and optimized to get into difficult to reach. The parachute is automatically detached to keep the position when the probe is completely landed. Electronics consist of 3 circuit boards used for detector operation, probe control and data transmission. Considering the payload capacity of a drone, totally 9 probes are loaded on a drone and spread out over the suspicious area. The probe is then collect radiological information for 15 min to locate and quantify gamma radiation level within area. Based on the geological and radiological information collected by each probe, the exact position of the source can be also calculated since the strength of radiation signals is proportionally changed by the distance from the source. The entire system is controlled by the O/S software developed by the present authors. The future step will be to try to optimize the experimental parameters in order to improve performance and accuracy.

Fig. 1. The schematic design of a gamma probe (left) and O/S software (right)

Table 1. Parametric values of a gamma probe

Parameter	Value
Physical Specifications	Length: 190 mm, Diameter: 80 mm, Weight: 600g
Sensor Type	CsI (12.5x12.5x20 mm)
Battery	Li-ion (0.7 V, 3500 mA)
Communication	GNSS, RoLA

***Keywords**: Disaster Response, Unmanned System, Radiation Mapping*

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PS6 (T6.3-0868)

Performance Evaluation of NaI- and HPGe-based Mobile Radiobioassay Laboratories for Individual Monitoring of Internal Contamination

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Intake of radionuclides can cause internal contamination. Whole body counting is a commonly used method to perform individual monitoring of internal contamination for high energy gamma-ray emitting radionuclides. However, in many cases, subjects who may have been exposed are located far from fixed whole body counting facilities. Therefore, it can be difficult to provide immediate and convenient monitoring by stationary systems. On the other hand, mobile whole body counting systems can provide not only rapid monitoring but also convenience to subjects. National Radiation Emergency Medical Center (NREMC) of Korea Institute of Radiological and Medical Sciences (KIRAMS) has been operating a mobile whole body counter, named Mobile Radiobioassay Laboratory (MRL), which was equipped with two large NaI detectors ($7.6 \times 12.7 \times 40.6$ cm^3 each) to perform individual monitoring since 2016. In addition, NREMC newly developed HPGe-based MRL to precisely estimate internal contamination of public and workers. Performance of the HPGe-based MRL was evaluated and compared with the previous unit in this paper.

The HPGe-based MRL was equipped with broad energy germanium detector (Model BE5030, Mirion technologies). It is a HPGe detector with relative efficiency of approximately 48 % and FWHM of 2.0 keV at 1.332 MeV. Energy range from 3 keV to 3 MeV can be covered by using this detector. Table 1 lists characteristics of two MRLs of KIRAMS. Measurement time required by HPGe-based MRL is longer than NaI-based MRL because of difference in efficiency. However, HPGe-based MRL has better energy resolution to identify radionuclides. Minimum Detectable Activity (MDA) of both MRLs is low enough to estimate internal dose below 1 mSv for intake of ^{137}Cs.

Table 1. Comparison of the characteristics of NaI- and HPGe-based MRLs

	NaI-based MRL	HPGe-based MRL
Type	Stand type	Bed type
Measurement time	180 s	1800 s
FWHM at 1.332 MeV	< 100 keV	< 2 keV
MDA	200 - 300 Bq	200 - 250 Bq
Throughput per 8 h workday	96 people	12 people

Keywords: *Mobile laboratories, Internal contamination, Individual monitoring*

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PS6 (T6.5-0934)

Radiation Dose Assessment for Recovery Workers of Contaminated Lands

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Deposited radioactive materials after a radiological or nuclear emergency result in exposures of the public in the intermediate- and long-term. Therefore, remedial action should be taken on contaminated lands for the evacuated residents to return to their homes and normal lifestyle. However, since the workers engaged in site cleanup exposed to radiation through various pathways, occupational dose assessment is necessary to optimize the occupational radiation protection. This study estimated the exposure dose to workers restoring the areas contaminated with Sr-90, Cs-134, Cs-137, and then calculated the maximum workable soil concentration to comply with the annual dose constraint of 20 mSv. In addition, the LHS (Latin Hypercube Sampling) – PRCC (Partial Rank Correlation Coefficient) sensitivity analysis was performed to identify the influence of the input parameters and their variation on the model outcomes. For the realistic assessment, the Korean characteristic data, the detailed exposure scenarios depending on the type of work, and the relevant exposure pathways were used. As a result, the most severe exposure-induced work type was identified as the excavator operation with an annual individual dose of 4.75E-01 mSv at the unit soil concentration (1 Bq/g), from which the derived maximum workable soil concentration was 4.21E+01 Bq/g. Dose contribution by radioisotopes decreased in the order of Cs-134, Cs-137, and Sr-90 on the assumption that those multiple isotopes uniformly exist inside the soil. Dose contribution by exposure pathways decreased in the following order: ground-shine, soil ingestion, dust inhalation, and skin contamination. Furthermore, the high sensitive model input parameters and their PRCC were found to be as the dilution factor (0.75) and as the exposure time (0.63). In conclusion, the results are expected to contribute to optimize radiation protection for recovery workers and to establish appropriate response procedures to be applicable in areas with high deposition density after a radiological or nuclear emergency.

Keywords: *Remedial action, Occupational radiation protection, Sensitivity analysis*

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PS6 (T6.5-1081)

Radioactive Waste Classification with Deep Learning to Minimize Radiation Exposure

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In recent years, in both the radiological protection and radioactive waste management communities, there has been increased attention on how to effectively manage non-power related nuclear waste. All countries have to manage radioactive waste generated by activities unrelated to the production of nuclear energy, including: national laboratory and university research activities; used and lost industrial gauges and radiography sources; and nuclear medicine activities at hospitals. Although much of this waste is not long-lived, the variety of the sources makes any general assessment of physical or radiological characteristics difficult [1].

Object detection is one of the most fascinating technology in the modern world [2]. Using an intelligent object identification program in radioactive waste classification is an advantageous approach when compared to the traditional classification method which is based on human goodwill and labor, due to a large number of objects that are identified in a shorter period of time [3-4], thereby minimizing the exposure time to radiation.

In this paper, we propose a deep-learning-based visual recognition and classification approach for radioactive wastes. We developed a program for the visual recognition and classification system mainly for combustible wastes, of which our focus lies on plastic vinyl/rubber/paper. Our system shows accuracy level of 98% (on 50th gen) for recognition and classification. The classification of the radioactive waste will be faster and intelligent using the proposed program without or reducing human involvement.

Keywords: *Radioactive waste, Classification, Deep learning*

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PS6 (6.B-0422)

Survey of Activities of Some Japanese Academic Societies toward the Public before and after the Fukushima Nuclear Accident

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After the Fukushima nuclear accident, the Japanese public paid considerable attention to messages from professionals and academic societies associated with radiation and risk. There have been a variety of types of communications between the public and professionals in terms of media, contents, phase, etc. In general, it is said that information and messages from professionals depended on their background and expertise, e.g. working in nuclear, medical or basic science sector, and being dedicated to radiobiology, radiation physics or radiation protection. The analysis of data on what, when and how such professionals have done (directly or indirectly) toward the public would allow us to have argument about possible approaches for risk communication, not only in an emergency exposure situation but also in an existing exposure situation. The aim of the present study was to compile and briefly review the transition of social activities of some radiation- or risk-associated academic societies before and after the Fukushima nuclear accident.

Activity records that were uploaded on the websites of six Japanese academic societies, including Japan Health Physics Society, were collected. The information was then organized according to objective indices such as media (e.g. website, lecture, face-to-face meeting), contents (e.g. radiation, risk, disaster, Q&A, consultation), target group (e.g. the general public, teacher, (local) government staff, professional), frequency, and year. The trends and characteristics of those activities are now being analyzed in terms of the aforementioned indices. Apparently, many academic societies did not have communications with the public before the Fukushima nuclear accident (in their normal activities), even though its importance was recognized to some degree. Also, the academic societies have been working on different types of social activities and communications with the public after the accident. Each academic society has its own perspectives, ideas and resources as well as strong points, which may be factors to determine how to be engaged with society.

The present study is based only on the survey of the facts or objective data. Possible and practical approaches for risk communication can be investigated more deeply, if we can also know how the public appreciated the communication with professionals. Further studies would be expected to provide an academic society with suggestions about its roles in and needs from society.

Keywords: *Academic society, General public, Risk communication*

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PS6 (6.B-1196)

Does Correct Knowledge Make People Feel Ease of Anxiety about Radiation Effects on the Next Generations after FDNPP Accident?

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It was thought that fear of the next-generations impact is one of the major psychological burdens in Fukushima after the FDNPP accident. Searching for ques to make them feel the ease of excessive anxiety, we analyzed data from a questionnaire survey by mail for residents of Fukushima prefecture aged from the '20s to '70s which was conducted in August 2016. The response rate was 43.4%, and 861 people sent back their answers. In this survey, 839 people replied about their anxiety related to radiation effects on the next generations.

Based on the fact that the better understandings of radiation effect is often one of the aims in risk communication for the public[1], we focused on the relationship between the level of anxiety and the correctness of the answer to five true/false questions: 1. the radiation remains in a body if exposed once. 2. the world standard of the concept for radiation safety is established based on the idea that the cancer mortality risk becomes more significant if you are exposed by more radiation dose. 3. there is no evidence of the radiation effects on the next generation in the study of atomic bomb survivors. 4. the DNA damage in the exposed cell by radiation can not be repaired. 5. The Japanese government sets the standard limits of radionuclide for food 100Bq/kg. The answers were true/false/not sure.

As a result of the linear regression analysis applying the relationship between the number of correct answers, and a degree of anxiety, the increasing of correctness and decreasing of incorrectness reduced the level of anxiety. Still, the number of "unsure" did not affect it. The contribution of explanation power was about 11%. To evaluate the effect of each question, we performed a qualification type 1 analysis. To answer correctly on question #1 and #3 makes anxiety less but incorrectly makes answerers feel more anxiety. The effect of question #2 was opposite, and the other questions did not affect it.

These results imply that having correct knowledge may reduce the anxiety of the radiation effects on the next generation, but in detail, some information may lead to confusion. We should pay attention to it under risk communication.

Keywords: *radiation anxiety, questionnaire survey, knowledge, next generation*

ACKNOWLEDGMENTS
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PS6 (6.C-0378)

Innovations In Training for Emergency Response

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Radiation regulations dictate that emergency response training is undertaken regularly. Such 'live-action' training can be very costly to undertake and if not organised correctly, can fail to achieve its goals. Training innovations, such as 'chess sets', interactive games, virtual reality and other computational solutions are being developed to enhance the user learning experience. Accessibility to on-line learning packages is also being modernised. These developments are aimed to minimise costs, embrace diversity and learning methods and to improve the efficacy of the learning experience.

PS7 (T7.1-0080)

Determining the Radon Emanation Coefficient for Soil Samples

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Exposure to high concentrations of radon and radon decay products are known to increase lung cancer in humans [1]. In order to protect the public and comply with international standards, the National Nuclear Regulator, through its Centre for Nuclear Safety & Security, has embarked on the development of comprehensive national radon research programme to determine possible issues of indoor radon exposure in the country. As an initial phase, CNSS and its partner institutions are in the process of designing a national radon survey strategy. The aim of this study is to determine the radon emanation coefficient for soil samples collected in various areas in South Africa. The radon emanation coefficient is important for identifying areas with high radium concentrations which emanate radon which could lead to high levels of radon in the houses [2]. The areas of interest for this study include Paarl and Saldanha with granite outcrops, Kloof, and Boksburg mine dumps where high radon levels are expected. The soil samples were collected and analyzed in the laboratory. Both alpha and gamma spectrometry was used for determining the radium activity of the soil samples and the radon released by the soil from which the radon emanation coefficient can be found [3]. The samples were sealed for a period of 21 days to achieve secular equilibrium between radium and radon and then counted using a NaI detector to measure the radium activity using a ^{214}Bi photo-peak [4]. The radon concentration was determined using an active radon monitor, the RAD7 which measures the alpha particles emitted from the radon progenies, ^{218}Po and ^{214}Po. The values obtained for the radium activity varied in the range of 94-771, 339–368, 64-147, 18-34 and 11 Bq.kg^{-1} in Boksburg, Kloof, Paarl, Saldanha, and Centurion samples, respectively. The radon concentration values built up in the accumulation chamber (of 10 litres) to values of 175-596, 225-352, 218-223, 61-99 and 58 Bq.m^{-3} in the Boksburg, Kloof, Paarl, Saldanha, and Centurion soil samples, respectively. Finally, the emanation coefficient was determined as the fraction of the radon concentration and radium activity, the values obtained from the highest to the lowest are 0.82, 0.28–0.87, 0.21- 0.54, 0.6- 0.31 and 0.10-0.15 in Centurion, Saldanha, Paarl, Boksburg and Kloof, respectively. These values of the emanation coefficient are in line with the typical values of 0.01- 0.8 reported in the UNSCEAR report [5]. These values will help to choose the distribution of houses that should be measured in the planned survey of radon in houses.

Keywords: *Radon, Radium, Emanation coefficient, Gamma spectroscopy, Alpha spectroscopy*

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PS7 (T7.1-0455)

Investigation of Meteorological, Diurnal and Seasonal Influences on Indoor Radon in Abeokuta, Nigeria

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Radon concentration and meteorological parameters, namely indoor air temperature, relative humidity and barometric pressure, were monitored hourly from May 2012 to May 2017 in a masonry building in Abeokuta, Nigeria, based on continuous detection of the alpha particles emitted by diffused radon gas and its decay products. The results showed diurnal variation with higher ^{222}Rn concentrations during night and early morning hours and lower values during afternoon hours. Correlations between radon concentration and the meteorological parameters were also observed and analyzed statistically. Radon concentration correlates positively with relative humidity (+0.4836), and negatively with both air temperature (-0.3757) and barometric pressure (-0.3463), but daytime radon concentration and air temperature correlate positively (+0.5951). The monthly averages of ^{222}Rn concentration range from <10 to 90 Bq m^{-3}, with the highest monthly averages occurring within the raining season (June – September). There is need for Seasonal Concentration Factors (SCFs) correction for measurements periods less than twelve calendar months.

Table 1. Test for correlation between radon concentrations and the three meteorological parameters

	Radon Conc.	*Temperature*	*Rel. Humidity*	*Pressure*
Radon Conc.	1			
Temperature	-0.375695031	1		
Rel. Humidity	0.483579651	-0.838840202	1	
Pressure	-0.346334346	-0.051685211	-0.295719083	1

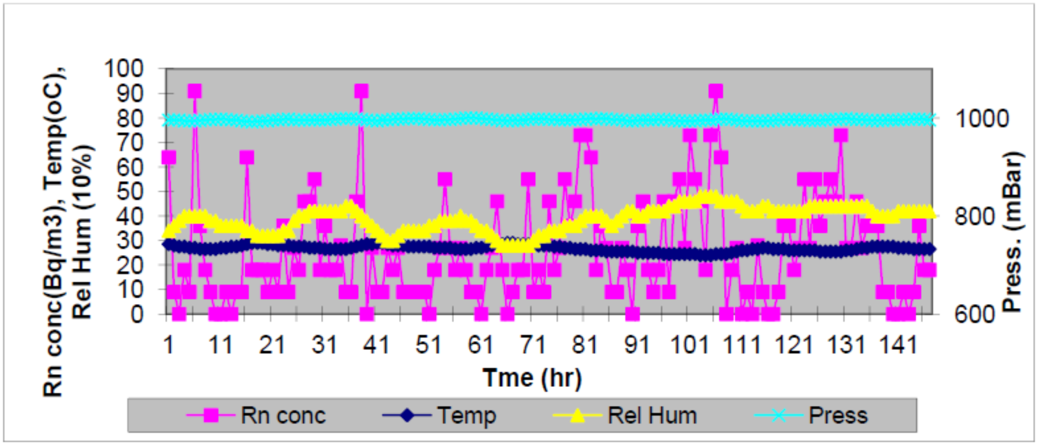

Figure 1. Correlation between radon concentration and the meteorological parameters

Keywords: *Indoor radon, Radon detection, diurnal and seasonal radon variation*

PS7 (T7.1-0572)

Introduction of a Modern and Easy Radon-in-Water Measurement Technology in Cuba

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The introduction in Cuba of a modern and easy technology for measuring the activity concentration of Radon in water is shown. It combines the use of the monitor "SARAD RTM1688-2" and the softwares "Radon Vision" and "Radon in Water Calculator", figure 1. Surface waters of rivers from Villa Clara, Cienfuegos and Sancti Spíritus provinces, and underground waters of aquifers Damují, Arimao from Cienfuegos, and Hanabanilla from Villa Clara at central Cuba were analyzed. Radon was low in rivers surface waters (0,010BqL^{-1}-0,17BqL^{-1}), and relative high in underground waters (0,060BqL^{-1}-101BqL^{-1}), table 1. The highest Radon values were found in Arimao aquifer, in relation with the presence of granites in its geology. The parametric value of 100BqL^{-1} for human consumption was not overcome neither in Damují nor in Hanabanilla aquifers. It was found a value of 101BqL^{-1} in Arimao aquifer, however, it came from a well not dedicated to human consumption. This situation leads to get deep in future researches in this area.

Table 1. Measured activity concentration of Radon in water

Sampling site	Average(BqL^{-1})	Uncertainty(k=2, BqL^{-1})	Minimum(BqL^{-1})	Maximum(BqL^{-1})
Rivers surface water	0.055	±0.012	0.010	0.17
Hanabanilla aquifer	0.306	±0.013	0.060	0.91
Damují aquifer	3.524	±0.071	0.34	12
Arimao aquifer	37.19	±0.39	5.4	101

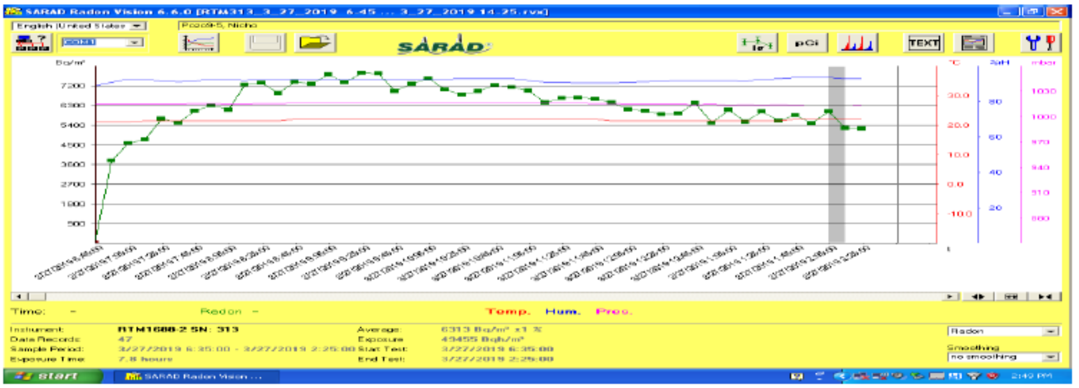

Fig. 1. Typical Radon-in-water temporal spectrum

Keywords: Radon measurement, water, Cuba

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PS7 (T7.1-0576)

Historic Overview of the Radiation Protection Policy for the Protection against the Dangers of Indoor Exposure to Radon in Dwellings in The Netherlands

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In The Netherlands indoor radon exposure was already recognised as an exposure risk in the 1980's and was, therefore, included in the Dutch radiation protection policy programme. In a national survey an average indoor radon concentration of 20 Bq/m^3 was measured in dwellings. Values above 200 Bq/m^3 were not recorded. Whereas the average indoor radon concentration is relatively low compared to the average values measured worldwide, radon was regarded as an exposure risk as the indoor radon concentration was about 5 times higher compared to the ambient radon concentration. In 1994 a radon policy statement was sent to parliament by the Minister of Housing, Spatial Planning and the Environment (VROM). This policy aimed to maintain the relative favourable indoor radon conditions in dwellings (i.e. maintain an average indoor radon concentration of 20 Bq/m^3). Building materials were found to be the dominant radon source in Dutch dwellings constructed after 1984. In the following decade research on radon sources, prevention and mitigation measures was continued and a second indoor radon survey was conducted. The health risk of indoor radon exposure was confirmed in 2000 by the Dutch Health Council (Gezondheidsraad). During this period it became eventually clear that further reduction of indoor radon by regulating construction products could not be achieved in a realistic and cost-effective manner. Instead, a mutual agreement was reached in 2004 for a period of 10 years between the Minister of VROM and the construction product industry with the aim to prevent a rise of indoor radon during this period. In this "standstill-agreement" the Ministry of VROM committed itself to continue monitoring indoor radon through national radon surveys and to commission studies on radon prevention and mitigation strategies. The industry committed itself to produce construction products that would not lead to an increase of indoor exposure to ionising radiation in new dwellings. In existing dwellings ventilation of occupied spaces is considered to be best strategy to mitigate the risk of indoor radon exposure. As mitigation measures cannot be legally imposed, encouraging home owners to improve the overall indoor air quality through ventilation is regarded as a suitable strategy. In the national radon survey of 2013-2014 indoor radon was measured in 2,500 dwellings. An average indoor radon concentration of 16 Bq/m^3 was measured. Some regional increase, up to an average value of 40 Bq/m^3, was observed in South Limburg and this variation can be explained by local geological conditions. This survey demonstrated that the standstill objective was maintained during the 10 year agreement period and that the national reference level of 100 Bq/m^3 was hardly exceeded. Based on the outcome of the survey, the Minister of Infrastructure and the Environment decided in 2015 not to continue the agreement. However, the established radon policy for new and existing dwellings will be continued. The knowledge and lessons learned of the past will be important ingredients for the Dutch national radon action plan.

PS7 (T7.1-0608)

Abnormal Radon Releasing before Earthquake and Its Effect on Ionosphere: An Overview and Challenges

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Abstract: Radon in the rocks is released with the activity of seismic fault belt and transported from underground to surface during the process of seismogenic. It is used as a precursor of great earthquakes by seismologists that abnormal radon appears above the fault zone. The phenomenon of ionospheric disturbance has also been observed before great earthquakes. It could be considered that there is some kind of relationship between abnormal radon and ionospheric disturbance before earthquake. Through the investigation of relevant research of the relationship between abnormal radon and ionosphere disturbance, we came to a preliminary conclusion of the relationship between abnormal radon and ionospheric disturbance and introduced the applications of earthquake prediction. However, it is difficult to verify the relationship directly by observation data presently. Hence, we proposed a simulation experiment scheme to verify the relationship.

Keywords: Earthquake, Abnormal Radon, Ionosphere Disturbance

PS7 (T7.1-0629)

Comparison of Different Methods of Measuring the Radon Equilibrium Factor

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The radon equilibrium factor is an important parameter for determining the inhalation dose of radon and its descendants. Its value remains between 0 and 1. There is a general agreement at present on the mean value of this factor to be assumed for domestic environments 0.4 (ICRP 65) , but it depends on several environmental parameters, so it can vary over a wide range. In this work, the equilibrium factor has been measured and compared using different methodologies.

The first methodology that has been studied consists on using a continuous air meter which measures the alpha and beta particles deposited in a filter where the descendants of radon are deposited. The filter is analyzed by gamma spectrometry and results allow to calculate the equilibrium factor Two different equipment has been tested. On the one hand the SmartCAM (Ultra Electronics), and on the second hand the air pump F&J DF-14ME.

The second technique is to expose several trace detectors, half of them with its chamber, that lets only radon through, and the other half with the nude detector material without a chamber, directly exposed to the air. The comparison between them gives the value of the equilibrium factor.

Equilibrium factor in constant laboratory conditions has been calculated and compared showing a good agreement.

Keywords: *radon, equilibrium factor, radon progeny*

ACKNOWLEDGMENTS

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Poster Session T7

PT7 (T7.1-071)

Radon Levels and Associated Health Impacts in the Radiation Contaminated Region in the Witwatersrand Area, South Africa

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Radon is known as the major source of human exposure to ionizing radiation. Epidemiological studies performed indicated an increased lung cancer risk with prolonged radon exposure. Several national authorisations implemented radon monitoring and established various radon mapping techniques with intentions to reduce exposure levels and protect the public. However, in South Africa there is no regulatory framework for the protection of the public against the risk associated with radon whereas there are areas which host uranium-bearing deposits and residues, which could serve as a significant source of radon. In addition, a large number of populations is living in close proximity to mine tailings. A recent small-scale study conducted in some parts of the Witwatersrand area, showed elevated radon levels and effective doses that could result to high lung cancer risks to the populations residing nearby tailings [1]. Therefore, the purpose of the present study is to measure the radon levels on a large scale and identify radon prone areas in the Witwatersrand area. Also, evaluate the magnitude of lung cancer risk linked to radon exposure in the area. In this study the assessment of radioactive materials from different source-terms will be carried out in order to help in establishing the radon risk map. Subsequently, the radon levels will be measured in indoor environments and will be integrated with health data so that any correlation with the occurrence of lung cancer could be established. The projected outcome will then be of importance in the regulatory of radon exposures and assessment of its potential risk on human health. To a greater extend, the results will add to the design of national radon mapping study which is within the mandate of the National Nuclear Regulator in South Africa.

Keywords: *Lung cancer, tailings, radon*

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PS7 (T7.1-0693)

Influence of Type of Activated Carbon and Water Adsorbed by the Material on Adsorption Capacity of Radon: Preliminary Test

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Radon is a radioactive noble gas that comes from ^{238}U disintegration. Thanks to its gaseous properties, it emanates from the soil and building materials and can reach high concentrations in closed spaces. Radon hazard comes from alpha radiation and when disintegrates starts a chain of short-lived radioactive elements, which are also radioactive nuclides that can be lodged in the lining of the lungs damaging their cells and increasing the risk of cancer. Therefore it is needed to control and reduce the exposition to radon inhalation, reducing exhalation from its origin or its concentration in the air.

Techniques that can be used to reduce its concentration in the ambient are ventilation but also treatment or conditioning of the soil or the materials that exhale it. Canisters with activated carbon are a tool widely spread used to detect and measure radon exhaled into air. Activated carbon is an adsorbent which has proven its good adsorption capacity to remove organic pollutants from water and air. So this work will explore the potential of activated carbon as adsorbent material that could be used to filter radon polluted air or as a barrier to mitigate radon exhaled, analysing the influence of type of carbon and water adsorbed by the material on its radon adsorption rate.

Different kinds of activated carbons from vegetal (Tecnasa, Kemira, Cabot 1020AW, Cabot 1020EN) or mineral origin (Lignite) have been used and it has been analysed their radon adsorption capacity (Bq/g). Moreover, it has been studied the influence of water adsorbed on radon adsorption capacity by changing the water content retained in the activated carbon.

The experimental set-up consists of a deposit, at the bottom of which a *pitchblende stone* is placed. Over the stone, the deposit is filled with soil and on its surface, a paper template is used to settle 7 canisters in a fixed position. To desorb radon, water or any substances adsorbed, the canisters are placed in the oven at 110ºC for 8 hours and after that, are set in their fixed position in the experimental set-up. In each test, the canisters are filled with a type of activated carbon dry. After being exposed for 24 hours, the canisters are left for 3 hours to reach secular equilibrium and then the adsorbed radon is measured through its descendants by gamma spectrometry with a NaI detector. Influence of water adsorbed in the material on radon adsorption capacity is studied following the same procedure but with activated carbon previously steeped in distilled water for 1 minute and set in the oven at 70ºC to dry until the water content retained desired is reached.

Results show that for an optimum result of adsorption of radon, mineral carbons (Lignite) are preferred over vegetal origin ones (the best is Kemira). Water content of the adsorbent reduces radon adsorption capacity, so use dry material is always advised. These recommendations will be used as a reference to design new strategies to treat radon polluted air or to reduce radon concentration exhaled from soil.

Keywords: *Radon, Adsorption, Activated Carbon*

ACKNOWLEDGMENTS

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PS7 (T7.1-0899)

Study on Sampling Representativity and Radon Levels' Seasonality in the Framework of a Systematic Survey in an Italian University Campus

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In light of the upcoming transposition into national law of Council Directive 2013/59/Euratom [1] (EU_BSS in the following), Italian National Institute for Insurance against Accidents at Work and the Basic and Applied Sciences for Engineering Department of Sapienza – University of Rome have recently promoted a radon survey in workplaces of university buildings. The survey has interested 11 buildings of the same Department campus, all with similar building characteristics and grouped within less than 1 km^2.

The measurements have a duration of one solar year to evaluate the annual average of radon activity concentration in air, as required by the EU_BSS. Rooms sampled for radon measurements (i.e. the overall sample) have been split into two subsamples: for the first group measurements have been performed for 2 consecutive 6-mo periods, for the second one measurements have been going through 4 subsequent 3-mo periods. Censed rooms, among whom the overall sample has been picked, are more than three hundred workplaces whose intended use is always clearly identified: administration and professors' offices, research and educational laboratories, conference rooms and classrooms.

The representativity of the overall sample is evaluated with respect to all the workplaces pertaining to the campus by considering two distinct influencing parameters: floor level and intended use, the latter reflecting on occupancy patterns and habits. The representativity of the sub-sample chosen for 3-mo-lasting measurements is then evaluated with respect to both the overall sample and the whole university department campus. The experimental results are discussed taking into account, the floor, the building characteristics, the distribution within the building and the intended use of rooms. Besides, results from the survey are used to evaluate if the averaged indoor radon concentration in a certain season could be assumed as representative of a larger period (i.e. semester and/or 9-mo periods) for specific intended uses and floor levels.

***Keywords**: radon survey, workplaces, university buildings*

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PS7 (T7.1-0900)

Radon Spatial and Seasonal Variations in University's Buildings Located in an Italian Karst Region

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In the framework of a collaboration between INAIL and University of Salento (UniSalento in the following) a radon survey in 54 buildings belonging to the UniSalento's campus was performed. The buildings are located mostly in a restricted area presenting a morphology characterized principally by marls, calcareous marls and calcarenites belonging to several Pleistocene sedimentary cycles (karst area).

The survey was performed monitoring for two consecutive semesters corresponding to spring/summer (SS) and autumn/winter (AW) about 1100 rooms located at different floors (see tab. 1). Moreover, in a restricted sample of about 250 rooms another radon monitoring was performed for six-months corresponding to winter-spring season (WS) in order to better characterize radon seasonal variations and to evaluate if the indoor radon concentration in a certain season could be assumed as representative of a one year radon level.

Table 1. Spatial distribution of the monitored rooms

Floor	N. of rooms
Ground floor	668
Basement	145
1st floor	197
2nd and upper floors*	90

For radon monitoring, passive devices with SSNTD were used; more information about laboratory technique and procedures are given elsewhere (1–3).

The analysis of radon level distribution respect to floors highlighted not negligible radon levels also in rooms located at upper floors, in particular in historical buildings (figure 1). This situation requires particular attention in choosing and testing the proper remedial actions.

The estimated annual Radon concentration are very similar to the ones calculated for the WS period (annaul_Rn/ WS_Rn =1.1). Regarding the ratio between SS and AW an inverted seasonal factor was found out: AW/SS=0.85. This phenomenon was also found in other karst areas (4), although generally it is not common to find higher radon concentration during SS than in AW.

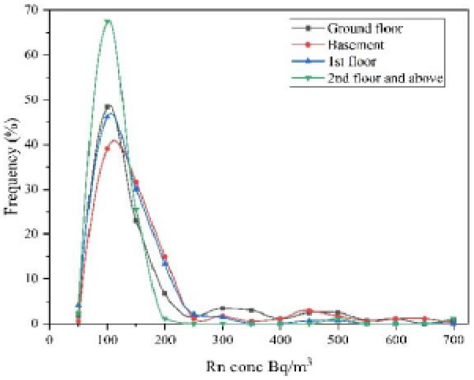

Fig. 1. Frequency distribution of annual average radon concentration at different floors

Keywords: Radon, seasonal variation, spatial distribution.

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PS7 (T7.1-0902)

A pomplots Analysis of AIRP International Radon In-field Intercomparisons

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According the ISO/IEC 17025:2017 (1) standard, the periodical participation in inter-laboratory comparisons (as Proficiency Test) is a tool laboratories may use to assess or to monitor their own performances. In the last years radon laboratories showed interest in testing their monitoring systems by in-field exercises, taking place outside of radon reference chambers in less controlled and more variable real conditions, similar to the ones in which radon passive devices are usually exposed. Therefore, a working group, leaded by AIRP, the Italian Radiation Protection Association, was established with the aim of organizing "radon passive detectors in-field intercomparisons".

The "International radon in-field intercomparison for passive measurement devices" is now in its third edition, all the exposures were performed in Italy and saw the participation of more than 80 laboratories.

The data analysis were performed according to the ISO 17043:2010 (2), for each radon level exposure test, the percent difference (PD), the normalized error (En) 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 (3). Moreover, both Youden plot and Mandel analysis were used to compare the exposure data for each intercomparison.

Recently, a further analysis has been carried out by using the "PomPlot method" (4), a graphical method suitable for the presentation of intercomparison results. This method allows to analyze simultaneously not only the results of the various exposures performed in the same intercomparison but also, data coming from different intercomparisons. This approach allows us to evaluate the performance of a single laboratory over time. In this work, the outcome of this activity is shown.

Keywords: *In-field intercomparison, radon, statistical analysis*

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PS7 (T7.1-0912)

Radon Concentration in Workplaces during Working Hours and Its Impact on Measurement Protocols: Results from an Italian Study

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Measurements of radon concentration in workplaces, whose results need to be compared with reference level (RL) adopted by national legislations, are generally performed using passive radon devices exposed for long periods (from 3 months up to one year). Since these devices integrate radon concentration both at night and in the daytime, they usually overestimate the actual radon concentration during working hours [1]. This could lead to incorrectly identify exceedances for some types of workplaces.

Some countries took this aspect into account in their measurement protocol performing – if annual average radon concentration exceeds RL – additional short-term measurements using active devices in particular types of workplaces, such as schools or public buildings served by mechanical ventilation systems [2][3]. Since active devices register radon levels at least every hour, they allow to estimate a correction factor to be applied to the results of long-term measurements in order to assess the actual concentration during working hours.

However, the above-mentioned protocols require that measurements using active devices are performed over one week (at maximum) assuming them as representative of a much longer period.

In order to verify this assumption, in the framework of an Italian study, an analysis was performed using data of radon concentration continuously measured in about 30 different Italian workplaces. For each of these workplaces, radon monitoring, using both active and passive devices, was performed over periods of at least 6 months. In this work, first results of this analysis are presented as well as their possible impact on measurement protocols for different types of workplaces.

Keywords: *radon in workplaces, working hours, radon measurement protocols*

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PS7 (T7.1-1006)

Some Considerations on Optimization, Reference Level and Action Level for Existing Exposure to Radon

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Protection from radon exposure, particularly in dwellings, has several peculiarities compared with the typical characteristics and related approaches of more "traditional" exposures to ionizing radiation.

Nevertheless, exposure to radon both in workplaces and in dwellings has been included in the general framework of radiation protection, as an existing exposure situation, in the last international and European Basic Safety Standards. For such situations, the main optimization tool is the Reference Level (RL).

Optimization should be applied with priority for initial radon concentration levels above the RL, but it should be applied also for initial radon levels below the RL.

However, in many national regulations the RL is used in the same way as the previous tool Action Level (AL), i.e. for levels below the RL no optimizing action is required.

On the other side, there is no regulation or guideline dealing with criteria to identify the initial radon level below which optimization is no more required or useful.

In this presentation, requirements and recommendations by ICRP and International and European BSS on optimization of radon exposure in workplaces and dwellings will be summarized and some examples of optimization below RL will be presented and discussed.

Keywords: *Radon, Optimization, Reference Level*

PS7 (T7.1-1020)

Indoor Radon Concentration in Dwellings Located at Limestone Zones in Korea

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Radon is well known to be the most significant contributor to human exposure from natural sources. IAEA recommended that national authority should determine radon exposure levels of the population by undertaking national and regional surveys (IAEA, 2013). In Korea, Korea Institute of Nuclear Safety (KINS) has conducted detailed sample surveys on 10 counties since 2015, considered as "Radon Priority Areas" based on the previous nationwide surveys in 1989 to 2009. More than 3% of total detached houses in each county were selected as samples with each distance over 500 m. The passive type alpha track detectors (Raduet®, Hungary) were installed in the living room or bedroom in the first floor for each house and replaced quarterly to obtain seasonal data. The annual mean indoor radon concentration for 10 counties was 126 Bq m^{-3} (66 – 150 Bq m^{-3}), twice as high as the national average 62.1 Bq m^{-3} (Kim et al., 2011). Among the counties, Yeongwol, Pyeongchang, Jeongseon, Danyang, and Jecheon were the large cluster of relatively high radon concentration with mean values of 150, 149, 143, 140, and 136 Bq m^{-3}, respectively. Geologically, these 5 counties were located in a representative large limestone zones in Korea. Paleozoic limestone, composed of the Choseon Supergroup, normally contains low ^{238}U concentration similar with an average of the earth's crust materials (Hakl et al., 1997). However, karst topography or caves formed from the dissolution of limestone could increase the overall permeability of the rocks and convective flow of radon may be induced (Gundersen et al., 1992). In fact, Danyang and Yeongwol counties are well known for karst topography and the distribution of large limestone caves. In a word, karst topography of limestone zones contributed significantly to the transport of radon gas, which was the main factor of the elevated indoor radon concentration in these areas.

Table 1. Indoor radon concentration in dwellings in Radon Priority Area

Location		Number of houses	Annual average (Bq m^{-3})		% > 148 Bq m^{-3}	% > 300 Bq m^{-3}
Province	County		AM	GM		
Gangwon	Yeongwol	481	150 ± 135	120 ± 1.8	29.5%	8.2%
	Jeongseon	484	143 ± 196	101 ± 2.1	24.9%	8.1%
	Pyengchang	452	149 ± 137	113 ± 2.0	32.5%	9.7%
Gyeongbuk	Munkyeong	610	102 ± 102	74 ± 2.1	17.1%	5.6%
	Yeongyang	83	66 ± 55	53 ± 1.9	7.5%	1.3%
	Uljin	432	105 ± 122	80 ± 1.9	14.5%	3.7%
Chungbuk	Geosan	512	107 ± 100	85 ± 1.9	16.3%	3.9%
	Danyang	360	140 ± 162	96 ± 2.2	26.9%	9.7%
	Jecheon	660	136 ± 127	105 ± 1.9	26.1%	7.4%
	Chungju	617	117 ± 105	96 ± 1.8	17.3%	5.2%
Total		4691	126 ± 133	94 ± 2.0	24.1%	6.6%

Keywords: *indoor radon, limestone, karst topography*

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PS7 (T7.1-1122)

Justification for the Design and Implementation of a National Program in Colombia for the Prevention and Mitigation of Medium and High Concentrations of Radon Indoors

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By 2018, in both sexes and at all ages, lung cancer produced the highest number of deaths in the world among all cancers with 2,093,876 (11.6%) deaths, while Colombia was the fifth cause of Death from cancer with 5,856 (5.7%) deaths. In 1988 the IARC classified radon-222 into group I of carcinogenic factors for humans, for its part, the World Health Organization (WHO) said that after tobacco, radon is the second most important cause of lung cancer, estimating that between 3% and 14% of cases of lung cancer are attributable to radon and it is recommended not to inhabit spaces that exceed the concentration of this gas in 100 Bq/m^3.

In Colombia, studies have been conducted outdoors, for seismic, volcanic and geothermal purposes and high levels of radon from both Torio-232, and uranium-238 were evidenced, reaching maximum emanation values between 333000 Bq/m^3 and 555000 Bq/m^3, levels far superior to those recommended by the WHO for interiors. Given that in Colombia radon concentration measurements have been made in outdoor spaces and close to geological faults, but the levels within the homes located around these sites are unknown, it is urgent to determine these levels indoors as a starting point that allows, In the medium term, generate a national radon program that includes prevention and mitigation strategies in places where concentrations exceed international reference values.

In 2015, WHO published a document with the guidelines that countries should take into account when designing and implementing a national radon program and in 2017 requested information, through a survey, from the Ministry of Health and Social Protection and the Ministry of Mines and Energy of Colombia of radon concentration levels indoors as well as local, regional and national government policies to prevent and mitigate radon concentrations above recommended levels. As there was no information to report to WHO, researchers of the National Institute of Cancerology E.S.E. and of the Colombian Geological Service were given the task of justifying the national authorities the need to determine the levels of radon concentration and from these levels design and implement a national radon program. Through working groups with experts and review of the specialized literature, these are the main findings that justify a national radon plan for Colombia:

Table 1. Factors that justify the implementation of a national radon program.

Measurements of radon concentration levels outdoors on average of 444000 Bq/m^3.
Populations located near places where radon concentration levels are much higher than those recommended by WHO.
Many small and intermediate cities in Colombia are built on geological faults.
Homes built without soil insulation.
Homes built with poor ventilation.
Homes built with materials that favor radon concentration.
In Colombian, the building construction regulations, radon is not considered a risk factor for health.
No measurements of radon concentrations have been made in underground and old buildings where people stay on average eight hours working.

Keywords: *radon concentration, lung cancer, building construction regulations.*

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PS7 (T7.1-1138)

Indoor Thoron Concentrations in Radon Priority Areas of Korea

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Thoron (^{220}Rn) is part of the ^{232}Th decay series, and is studied considerably less than its counterpart ^{222}Rn as it was assumed that the contribution from thoron's decay products to the total exposure from radon isotopes is mostly limited to around 10% (UNSCEAR, 2006; De With at al, 2018). Nevertheless, some studies have demonstrated that in certain cases exposure to thoron progeny in dwellings may be significant (De With at al, 2018). In Korea, Korea Institute of Nuclear Safety (KINS) has conducted surveys on 10 counties considered as radon priority areas based on the previous nationwide surveys in 1989 to 2009. The passive type alpha track detectors (Raduet®, Hungary) were installed closed on the wall in the living room or bedroom in the first floor. The annually arithmetic and geometric mean of indoor thoron concentrations for 10 counties were 65 ± 117 Bq/m^3 and 36 ± 2.8 Bq/m^3, respectively. The concentration of indoor thoron of the counties related in granite area (Munkyeong, Uljin and Goesan county) higher than those of the counties related in limestone area (Yeongwol, Jeonseon, Pyeongchang, Danyang and Jecheon county). The mean concentrations of thoron in old Korean-style houses built with mud were 138 ± 126 Bq/m^3, about two times higher than those in other style houses. The indoor thoron concentration in old Korean-style house seems to be correlated with the concentrations of ^{232}Th in surface soil and geological distribution of granite, because the old houses with high indoor thoron concentration were built with the building materials taken in the vicinity of the houses (Kim et al, 2007). Meanwhile, in case of no barrier against thoron such as mud tile and mud itself as finishing materials of the wall, the concentration of indoor thoron was much higher. For example, in the 26 houses used of mud tile without wallpaper, the mean concentrations of indoor thoron was 832 ± 482 Bq/m^3. Appropriate thoron barriers such as wall paper or paint with low air permeability should be used to reduce indoor thoron exposure.

Table 1. Indoor thoron concentrations in dwellings in Radon Concerned Area

Location		Number of houses	Annual average (Bq m^{-3})		Max	Remark
Province	County		AM	GM		
Gangwon	Yeongwol	437	52 ± 135	27 ± 2.5	1,569	Limestone
	Jeongseon	438	44 ± 76	27 ± 2.4	1,053	Limestone
	Pyengchang	421	59 ± 141	29 ± 2.8	1,955	Limestone/Granite
Gyeongbuk	Munkyeong	564	91 ± 130	51 ± 2.8	1,308	Granite
	Yeongyang	59	48 ± 164	20 ± 2.6	1,174	-
	Uljin	399	79 ± 114	45 ± 2.8	1,071	Granite
Chungbuk	Geosan	474	83 ± 134	50 ± 2.6	1,538	Granite
	Danyang	344	61 ± 134	32 ± 2.7	1,583	Limestone
	Jecheon	612	64 ± 96	36 ± 2.8	842	Limestone
	Chungju	567	51 ± 67	32 ± 2.5	925	Granite
Total		4305	65 ± 117	36 ± 2.8	1,955	

***Keywords**: indoor thoron, granite, thorium*

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PS7 (T7.1-1192)

Evaluation of Radon-induced Cellular DNA Damage in Lung Tissue based on Geant4-DNA Simulation

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The risk from exposure to inhaled-radon is of great concern in occupational and public health. Internal dose resulting from radon inhalation may eventually lead to lung cancer through cell mutation. Lung tissue comprises cells that layer at different depths in tissue. Exposed to alpha emissions of high linear energy transfer (LET) at the surface of lung epithelium, cells would show damage complexity that varies depending on the nucleus size and the depth in tissue.

This study investigated cellular damage from radon exposure by a Monte Carlo simulation of α-particle interactions with lung cells on the nanometer-scale. A segment of lung-tissue including ciliated and basal cells' nuclei, and the mucus layer was modelled. Geant4-DNA toolkit was used to simulate the nanometer-scale track structure of alpha particles in nucleus volume. Energy depositions along the track in cell nuclei were integrated into DNA damage through a damage-clustering algorithm. Subsequently, the distributions of nuclear dose, hit probability, accumulated DNA damage, and damage complexity were assessed at different levels of radon decay activity.

***Keywords**: Radon, Lung Epithelium Damage, Geant4-DNA*

PS7 (T7.1-1201)

Validation of Measurement Methodology for Radon Activity in Air and Water with a Liquid Scintillation Counting

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Radon is a well-known radioactive noble gas and can be detected everywhere. Many researchers have measured radon concentration as a means to assess lung cancer risk from the viewpoint of human health sciences. On the other hand, radon can be used as a radiotracer for predicting earthquakes and for understanding the environmental behaviors of atmospheric pollutants. Many measurement techniques for radon activity in the environment have been developed. A liquid scintillation counter (LSC) is generally used for the measurements of alpha and low-energy beta particles, and radon in water samples is generally measured using the LSC. Recently, we have commenced collaborative research with PerkinElmer Japan on validation of a measurement methodology for radon activity using a new type LSC (Quantulus GCT 6220; Fig. 1, left). In this study, for radon in water measurements, first we evaluated the minimum detectable concentrations (MDCs) using several measurement modes (normal, α/β, Guard Compensation Technology [1]) of the new type LSC. Well water samples were collected at the Kobe Pharmaceutical University campus for the evaluation. To validate the reliability of the measured radon concentration in water, the values obtained by the new type LSC were compared with the values obtained by the LSC of Kobe Pharmaceutical University, which was secondly calibrated with a standard ^{226}Ra solution. Next, radon is known to be physically adsorbed on activated charcoal. The activated charcoal collector (PICO-RAD^{TM}) is commercially used for radon gas measurements by applying this phenomenon. For the methodology, we have proposed to measure radon activity adsorbed on activated charcoal using the modes of new type LSC. Then second, in this study, we evaluated the elution time constant and conversion factor to radon concentration in air using a radon exposure chamber at Hirosaki University (Fig. 1, right).

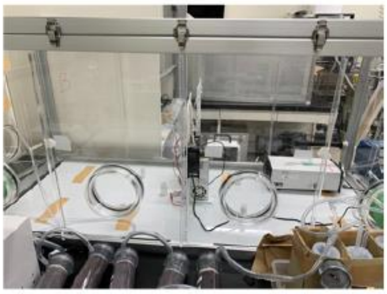

Fig. 1. Liquid scintillation counting system (left) and radon exposure chamber (right).

Keywords: *radon, liquid scintillation counter, activated charcoal*

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PS7 (T7.2-0150)

Natural Radiation Dose Assessment with the Cape Coast Metropolis OF GHANA

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Environmental gamma radiation was measured in the Cape Coast Metropolis to ascertain the radiation doses received by the inhabitant from natural sources. Measurements were carried out using a calibrated survey meter Rados 200 together with Geoexplorer 11 GPS, used to determine the locations of measurement points. Dose rates measured ranged from 0.056 μGyh-1 to 0.101 μGyh-1 with an average of 0.069 μGyh^{-1}.The highest dose rate of 0.101 μGyh-1 occurred at the University of Cape Coast and the minimum occurred at Ansapetu. The Effective Dose was calculated to be 0.343 mSv low at Ansapetu to a higher of 0.619 mSv at University of Cape Coast. The average population collective dose was calculated to range from 0.023 to 7.061 man-Sv. The difference in results is attributed to land setting, weather and cosmic ray intensity among others. The values are within the world average background dose rate of 57 $nGyh^{-1}$ in UNSCEAR (2000) report. The effective dose in the study area was within ICRP 60 recommended annual limit of 1 mSv for public exposure.

PS7 (T7.2-0419)

Radiological Implications and Dose Analysis of Primordial Radioactivity in Soil Samples of Himachal Pradesh, India

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The present study examined the natural radioactivity in soil samples of Solan and Shimla districts of Himachal Pradesh, India and their associated environmental impacts by using NaI(Tl) gamma-ray spectrometer. Radium equivalent activity (Ra_{eq}), dose rate, effective dose equivalent (EDE) for different age groups and doses to various organs, external hazard index (H_{ex}), internal hazard index (H_{in}) and radioactivity level indices were calculated and found well below the recommended criteria. Thus, the soil was considered safe for construction purposes. The average EDE for infants, children and adults was 0.95, 0.85 and 0.74 mSv y^{-1} respectively, which was greater than the world average values of 0.62, 0.55, 0.48 mSv y^{-1} reported by UNSCEAR, 2000. Organ-specific for different age groups were computed and received in the order Testes > Red Bone Marrow > Lungs > Ovaries. A good positive correlation was observed between radium content in soil and radon mass exhalation rate.

Table 1. Statistical variation of Specific Activity, Dose rate, hazard indices and ELCR

Parameter		Min	Max	Average	SD
Specific Activity ($Bq\ kg^{-1}$)	C_{Ra}	25	73	37	11
	C_{Th}	31	109	59	20
	C_K	215	701	430	140
	Ra_{eq}	105	270	152	44
Dose Rate ($nGy\ h^{-1}$)	D_{in}	93.10	234.05	133.34	38.10
	D_{out}	48.69	124.06	70.68	20.29
Radiological Hazard Indices	$(H_{ex})_I$	0.29	0.74	0.42	0.12
	$(H_{ex})_{II}$	0.14	0.37	0.21	0.06
	H_{in}	0.38	0.94	0.52	0.15
	Representative gamma risk, $I_{\gamma r}$	0.76	1.97	1.12	0.32
	Alpha index, I_α	0.12	0.36	0.18	0.06
	Activity Utilization Index, AUI	0.77	2.04	1.09	0.34
	Annual Gonadal Dose Equivalent, AGDE	262.09	698.31	374.50	115.36
ELCR x 10^{-3} (Sv^{-1})	Infants	2.32	5.85	3.33	0.95
	Children	2.07	5.20	2.96	0.85
	Adults	1.29	3.25	1.85	0.53

Keywords: *Gamma-ray spectroscopy, Primordial radionuclides, Hazard indices*

ACKNOWLEDGMENTS
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PS7 (T7.2-0738)

Natural Radioactivity In Soil And Water Used In Irrigated Farms In Lagos Metropolis

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Understanding the pathways through which radionuclides move in the environment makes it feasible to block or minimize man's exposure to radiation. The study focused on determining the sources of elevated concentrations of natural radionuclides in vegetables cultivated in different farms in Lagos metropolis. The activity concentrations of ^{226}Ra, ^{232}Th and ^{40}K in irrigation water and irrigated soil were determined using Hyper Pure Germanium (HPGe) detector.

The mean activity concentrations for ^{226}Ra, ^{232}Th and ^{40}K in irrigation water were 2.29 ± 0.28, 0.38 ± 0.03 and 2.17 ± 0.51 Bq/l. The overall mean concentration of ^{232}Th is 38 % of the reference value while the mean concentration of ^{226}Ra is 129 % higher than the WHO reference value for ^{226}Ra in water. The mean specific activity for ^{226}Ra, ^{232}Th and ^{40}K in irrigated soils were 10.99 ± 3.75, 11.20 ± 5.36 and 19.38 ± 15.81 Bq/kg. These are 30 %, 25 % and 19 % of the average world's values respectively.

The mean absorbed dose (D), annual effective dose value (indoor and outdoor) (E_{in} and E_{out}), radium equivalent (R_{eq}) from irrigated soils were 12.50 nGy/h, 61.38 µSv/yr, 15.35 µSv/yr and 28.4983 Bq/kg respectively. The values for the internal hazard index (H_{in}), external hazard index (H_{ex}), activity gamma index (I_γ) and excess lifetime cancer risks (ELCR) were 0.11, 0.078, 0.198 and 0.0537 x 10^{-3} respectively. The hazard indices D, E_{in}, E_{out}, R_{eq} H_{in} , H_{ex} , I_γ and ELCR were all below the world's average and safe limits.

Long-term use of irrigation water with elevated concentration of ^{226}Ra on vegetable farms could lead to excessive accumulation of ^{226}Ra in irrigated soil and consequently in cultivated vegetables. The study has therefore identified irrigation water as a more significant source of radiological contamination to irrigated vegetables in Lagos metropolis compared to irrigated soil.

Table 1. Activity concentrations of ^{226}R, ^{232}Th and ^{40}K and the WHO reference values.

Water Source	Activity concentrations (Bq/l)		
	^{226}R	^{232}Th	^{40}K
Deep well	1.85 ± 0.75	0.33 ± 0.09	2.92 ± 1.35
Surface water	2.42 ± 0.28	0.40 ± 0.03	1.9 ± 0.54
Overall mean	2.29 ± 0.28	0.38 ± 0.03	2.17 ± 0.51
Reference value	1.00	1.00	NA

NA –Not Applicable - No Reference value for ^{40}K

Keywords: *Natural Radioactivity, Irrigated soil, irrigation water*

PS7 (T7.2-0776)

Natural and Artificial Radioactivity in Volcanic Ash Soils of Jeju Island, Republic of Korea, and Assessment of the Radiation Hazards: Importance of Soil Properties

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In this study, we investigated the correlation between activity concentrations of natural (^{226}Ra, ^{232}Th, ^{40}K) and artificial (^{137}Cs) radionuclides and physical characteristics of soils in Jeju Island, and radiation hazard indices arising from the activity concentrations. Soil samples were collected from dark brown soils (DBS), very dark brown volcanic ash soils (VDBAS), and black volcanic ash soils (BVAS), based on the soil color. The average activity concentrations of ^{226}Ra, ^{232}Th, ^{40}K, and ^{137}Cs in all the soil samples were 32.4 ± 6.53 Bq kg^{-1}, 35.6 ± 7.68 Bq kg^{-1}, 314 ± 107 Bq kg^{-1}, and 20.8 ± 14.8 Bq kg^{-1}, respectively. The activity concentrations of ^{232}Th and ^{40}K based on soil color were in the order of BVAS < VDBAS < DBS, but ^{137}Cs was highest in BVAS, and ^{226}Ra was lowest in VDBAS. The activity concentrations of ^{137}Cs showed positive correlation with soil organic matter and the clay fraction and negative correlation with bulk density. In particular, ^{137}Cs correlated strongly in VDBAS and BVAS with high soil organic matter and clay contents. Comparison between the radionuclides showed that the correlations between the natural radionuclides were all positive, while ^{137}Cs, an artificial radionuclide, showed strong negative correlations with ^{232}Th and ^{40}K, respectively. The radiation hazard indices calculated from the activity concentrations were negligible with values lower or similar to global average values or those recommended by international organizations. In terms of annual outdoor effective dose rate (AEDR$_{out}$), the contribution of radionuclides to the soils was ^{137}Cs (5.1%) < ^{40}K (24.7%) < ^{226}Ra (29.0%) < ^{232}Th (41.3%), i.e., dominated by natural radionuclides, but contributions were dependent on soil properties.

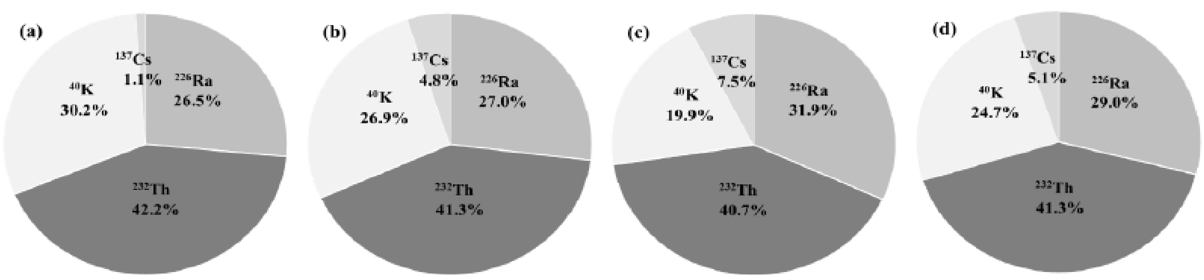

Fig. 1. Contribution of AEDR$_{out}$ values calculated from the activity concentrations of natural and artificial radionuclides in volcanic ash soils collected from Jeju Island (a: Dark brown soils, b: Very dark brown volcanic ash soils, c: Black volcanic ash soils, d: All soils)

Keywords: Natural radionuclides, ^{137}Cs, radiological hazard

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PS7 (T7.2-0961)

Risk Assessment due to ^{226}Ra, ^{232}Th and ^{40}K in Soil from Kilimambogo, Kenya

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The levels of naturally occurring radionuclides present in the environment vary from place to place depending on the rock formation and the chemical properties. The distribution of primordial radionuclides in soil samples from Kilimambogo region in Kenya were determined using Thallium doped Sodium Iodide NaI (TI) gamma spectrometry technique. Activity concentrations of ^{226}Ra, ^{232}Th and ^{40}K, ranged from 21 ± 5 to 103 ± 25 Bq kg^{-1}, 27 ± 2 to 101 ± 6 Bq kg^{-1} and 108 ± 2 to 1495 ± 32 Bq kg^{-1} with mean concentration levels of 45 ± 6 Bq kg^{-1}, 57 ± 5 Bq kg^{-1} and 603 ± 46 Bq kg^{-1} respectively. ^{226}Ra and ^{232}Th were within the world averages of 33 and 45 Bq Kg^{-1} respectively while ^{40}K exceeded the world average of 420 Bq Kg^{-1}. Radiological risk indices were determined in order to evaluate the radiation hazards from the soil. The mean external, H_{ext} and internal, H_{int} hazard indices were 0.46 ± 0.12 and 0.59 ± 0.14 respectively which were lower than the recommended value of 1. The mean annual outdoor and indoor effective dose equivalent were 0.20 ± 0.05 and 0.30 ± 0.07 mSv y^{-1} respectively. These values were higher than the world average values of 0.41 mSv y^{-1} and 0.07 mSv y^{-1} respectively but less than 1 mSv y^{-1} recommended by the International Commission on Radiological Protection (ICRP) for the public. Overall, an individual who lives in Kilimambogo area would receive an annual total effective dose ranging from 0.29 to 0.75 mSv y^{-1} with an average of 0.50 mSv y^{-1} from soil which is comparable to the world average value of 0.48 mSv y^{-1}. The results emanating from this study will be used by other researchers as a baseline for the radiation exposure of the residents in the region in future studies.

***Keywords**: Risk Assessment, Radionuclides, Kenya*

PS7 (T7.2-1114)

Temporal Variation of Medically-derived ^{131}I Concentration in Effluents of Sewage Treatment Plants and Its Impact on the Nearby River

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Recently, the use of ^{131}I for diagnosis and treatment of thyroid cancer has been increasing, and ^{131}I administered to patients is continuously discharged into the water environment after being flowed into sewage treatment plants (STPs) through hospitals and homes. In this study, the concentrations of medically-derived ^{131}I in effluents of STPs were investigated by hourly, daily, and seasonal to assess its impact on the nearby river. The survey sites were based on two STPs located in the mainstream of the Yeongsan River in Korea, and included four mainstreams located near these STPs. Analysis of ^{131}I radioactivity was measured using a gamma-ray spectrometer consisting of a high purity germanium detector and a multi-channel analyzer. Most of the ^{131}I nuclides in the STPs were detected in the effluents at the two sites, but they were in the range of <30.6–4304 mBq L^{-1} (STP-1) and <33.7–3676 mBq L^{-1} (STP-2), respectively, depending on the time of investigation and the characteristics of the STPs. For the temporal variation, the concentration of ^{131}I in the effluent tended to be high in winter only at the STP-1 site, and no clear trend was found. However, according to a survey after a long-term holiday, ^{131}I concentrations in the effluent were either undetected or relatively low, confirming that they were closed related to medical and human activities. As a results, it would be useful to evaluate the behavior of ^{131}I nuclides in rivers at an emergency radiation accident.

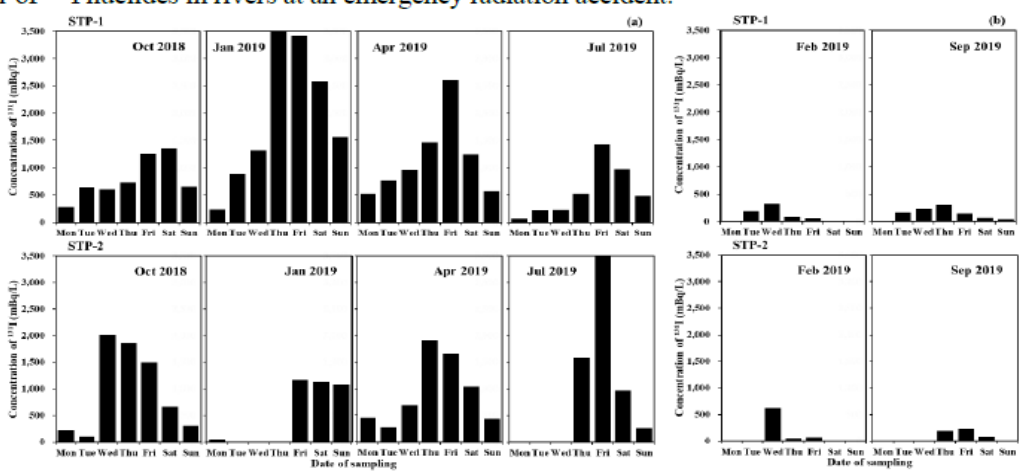

Fig. 1. Temporal variation of medically-derived ^{131}I concentration in effluents of STPs (a: seasonal, b: after long-term holiday)

Keywords: ^{131}I, medically-derived, sewage treatment plants

ACKNOWLEDGMENTS

This work was supported by a grant from the National Institute of Environment Research (NIER), funded by the Ministry of Environment (MOE) of the Republic of Korea (NIER-2019-03-01-013).

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PS7 (T7.2-1172)

Changes on Absorbed Dose Rate in Air Related with Road Improvements on Phu Quoc Island, Vietnam

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Road improvement programs utilize building materials containing natural radionuclides that will change absorbed dose rate in air. Phu Quoc Island (589 km^2), the largest island in Vietnam, is located in the westernmost area of Vietnam. In recent years, road improvements related to resort development have taken place on this island introducing the possibility that the absorbed dose rate in air and external exposure on the island have changed. In December 2015 before road improvements were undertaken, the present researchers determined absorbed dose rate in air on the island and found the median value of absorbed dose rate in air measured on asphalt or unpaved roads for the entire island was 37 nGy h^{-1} (n = 774) [1]. In the present study, a car-borne survey using a NaI(Tl) scintillation spectrometer was carried out on Phu Quoc Island in March 2020 and changes of dose rate in air before and after road improvements were estimated.

Roads which could be accessed by car were chosen for this survey and the route was approximately 150 km long. Mainly roads in the eastern part of the island had been upgraded from unpaved roads to concrete paved roads (approximately 30 km long) in the five years since the first study. The median value of absorbed dose rate in air for the whole island was 48 nGy h^{-1} (n = 638), and this resulted in about 30% higher dose rate compared to the rate before road improvement. Additionally, the median measured for the specific area of road improvement changed from 27 nGy h^{-1} for unpaved roads in 2015 (n = 321) to 34 nGy h^{-1} for concrete paved roads in 2020 (n = 74). Fig. 1 shows the cumulative frequency distributions of the absorbed dose rates measured for the whole island in 2015 and 2020, and about 50% of the measured data increased because of road improvement programs. The results showed that the absorbed dose rates in air were clearly changed by the road improvement, but it was assumed that there was no new radiation risk from the built-up environment for Phu Quoc Island.

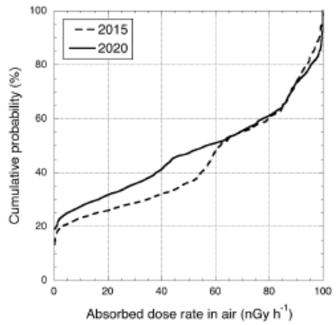

Fig. 1 Cumulative frequency distribution of absorbed dose rates in air measured in 2015 and 2020.

Keywords: *Absorbed dose rate in air, Car-borne survey, Vietnam*

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PS7 (T7.3-1094)

A Study on the Selection Schemes of the Appropriate Computer Programs for Evaluating Cosmic Radiation Exposure at Commercial Aircraft Altitude

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Recently, former Korean aircrews filed for the leukemia industrial accidents and it was reported in the media. This led to the social issue of exposure to cosmic radiation for aircrew members. the limitations of computer programs used for crew dose assessment by cosmic radiation were pointed out in the process of national audits, parliamentary debates, etc. Meanwhile, reassessment of the average exposure dose from cosmic radiation in Korea is necessary to reflect the increase in the number of international air routes and passengers, advanced cosmic radiation analysis technologies. Investigation, analysis and selection of computer programs are needed to be applied for reevaluation. In order to improve crew safety management and reassessing the national average exposure, "selection schemes of the appropriate computer programs", a common factor for both purposes, are studied in this research. Domestic and international trends in cosmic radiation dose assessment programs are analyzed and the appropriate programs are preliminarily selected for the purpose of evaluation.

The scope and contents of the study are as follows. First, domestic laws, regulations and evaluation programs on cosmic radiation safety management were investigated. Second, screening criteria for selecting the appropriate programs were set by analyzing the technical background documents of domestic laws and literatures of international organization and developed countries on cosmic radiation assessment. Third, suitability of evaluation programs was comprehensively analyzed against the criteria. Forth, computer programs for each evaluation purpose were preliminarily selected. Finally, the appropriateness of overall research process and results were verified and supplemented by the case studies on route dose assessment for each evaluation program.

As a result of analyzing the appropriate programs to confirm regulatory compliance, it is necessary to review the validity and adequacy of the programs presented in the current Korean safety guide. As a result of analyzing the programs suitable for the evaluation of national average dose, the appropriate programs for the purpose were preliminarily selected and will be applied to the evaluation of the average annual dose of cosmic radiation in Korea. The screening criteria derived from this study can be applied as a detailed analysis criteria of evaluation program for factual survey of airline companies.

Keywords: *Cosmic Radiation, Aircrew, Computational Dosimetry*

ACKNOWLEDGMENTS
This work was supported by the Nuclear Safety Research Program through the Korea Foundation of Nuclear Safety (KoFONS), granted financial resource from the Nuclear Safety and Security Commission (NSSC), Republic of Korea. [No.1803013]

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Poster Session T7

PS7 (T7.3-1095)

The Estimation of Annual Effective Dose due to Cosmic Ray at the Ground in Korea

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Everyone in the world is exposed to natural sources of ionizing radiation. The worldwide average exposure to cosmic rays contributes to about 17% of total annual effective dose(2.4mSv). To understand the population exposure to natural radiation, it is necessary to estimate the exposure of cosmic ray at the ground level.

In this study, the Korean population-weighted average annual effective dose due to cosmic ray was estimated. The formulas of UNSCEAR were used to estimate dose rates from photons, neutron and direct ionizing component of cosmic radiation at the ground level. It was also estimated that the effective dose rate due to cosmic rays considering the various values for longitude, latitude, water density in the ground and solar cycle influence. Both models, such as EXPACS and CARI-6, take all these variables into account. A mean shielding factor of 0.8 and an indoor occupancy factor 0.8 have been assumed. The nation's population statistics by administrative region used data from the National Statistical Office as of December 2017.

As a result, the amount of cosmic ray exposure received at the ground is about 0.39 mSv using the formulas of UNSCEAR, and 0.33 mSv considering the shielding and indoor occupancy factor. This is smaller than the worldwide average exposure (0.38 mSv) by UNSEAR, but it is bigger than the results using different models. The results of this study are expected to help Koreans understand the exposure to cosmic ray among the natural radiation.

Keywords: *Cosmic ray, Effective dose, UNSCEAR*

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This work was supported by the Nuclear Safety Research Program through the Korea Foundation Of Nuclear Safety (KoFONS), granted financial resource from the Nuclear Safety and Security Commission (NSSC), Republic of Korea. (No. 1803013)

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PS7 (T7.4-0073)

Assessment of NORM in Fruits and Vegetables from Hartbeespoort, Mafikeng and Pretoria Markets

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This study aims to measure the activity concentration in fruits and vegetables collected from local markets in Hartbeespoort, Mafikeng and Pretoria. The gross alpha and beta activity concentration in fruits and vegetables collected from Hartbeespoort, Mahikeng, and Pretoria have been evaluated using Oxford proportional gas-flow series. A total of 4 fruits and 7 vegetables samples from these areas were analysed. Hartbeespoort, Mafikeng and Pretoria activity concentration of gross alpha and beta in fruits were 0.527 ± 0.182 Bq/g and 10.015 ± 0.576 Bq/g; 0.181 ± 0.053 Bq/g and 9.520 ± 1.449 Bq/g and 0.184 ± 0.006 Bq/g and 9.351 ± 2.281 Bq/g respectively. Hartbeespoort, Mafikeng and Pretoria sample results for gross alpha and beta activity in vegetables were 0.218 ± 0.150 Bq/g and 10.099 ± 2.435 Bq/g; 0.222 ± 0.137 Bq/g and 12.029 ± 1.718 Bq/g and 0.175 ± 0.191 Bq/g and 11.099 ± 2.238 Bq/g respectively. The highest gross alpha and beta activity in vegetables was found in Mahikeng and the highest gross alpha and beta activity in fruits was obtained in Hartbeespoort. The results obtained are higher than the WHO [3] standard limit of 1.0 Bq/g for gross beta and 0.1 Bq/g for gross alpha, however these results are comparable to the reported results in [1] and [2].

The neutron activation analysis (NNA) was used to identify naturally occurring radioactive material (NORM) by means of gamma spectrometry analysis using (HPGe) detector. The results showed that the highest average activity concentration in fruits were 1.015±0.627, 1.024±0.634, 0.049±0.025, 2.700±0.674 Bq/kg and the highest average activity concentration of ^{238}U, ^{234}U, ^{235}U and ^{232}Th in vegetables were 1.662±1.406, 1.674±1.419, 0.076±0.064, 4.228±1.302 Bq/kg and respectively. The levels of activity concentrations of ^{238}U, ^{234}U, ^{235}U and ^{232}Th are higher in Hartbeespoort followed by Mahikeng and Pretoria. The activity concentration of ^{238}U, ^{234}U, ^{235}U and ^{232}Th were lower than the world average values reported in UNSCEAR report of 2000 [4].

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PS7 (T7.4-0391)

Study on Radiation Environmental Impact of Exploitation and Utilization of Rare Earth Ore in China

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Taking Bayan Obo mining area of Baotou Iron and Steel Company as a typical case, the radiation environmental impact caused by the gas effluent emission from rare earth mining in China was investigated and evaluated. The evaluation results show that it has caused significant impact on the surrounding residents, although still lower than the limit of personal effective dose (1mSv/a). As to improving the local radiation environmental quality, some measures should be taken as follows. Suitable covering materials should be selected to cover the waste slag yards and dumps as soon as possible. The locations of the new waste slag yards and dump yards should be kept as far away from the urban area of Bayan Obo mining area as possible. The dust emission during mining and transportation should be further reduced. And it is also very important to strengthen the engineering ethics and radiation protection expertise of relevant personnel so as to optimize the planning and design of the development and utilization process.

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PS7 (T7.4-0568)

Detection and Characterization of Naturally Occurring Radioactive Materials in Industrial Products by Combining In-situ and In-lab Measurements

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The detection and characterization of NORM in scrap and abrasives, used by industries of central Cuba is shown. They were combined in-situ measurements of the environmental equivalent gamma dose rate, carried out with a verified portable radiometer, and in-lab high resolution gamma spectrometry analysis (Figure 1). The average of the environmental equivalent gamma dose rate was (0,43±0,02)μSvh^{-1}, minimum 0,01μSvh^{-1}, maximum 2,07μSvh^{-1}, four times higher than the normal background(0,09±0,02)μSvh^{-1}. They were determined relatively high values of Tl-208, Pb-212, Bi-212 and Ac-228, as well as Pb-214, Bi-214 and Th-234, from the Th-232 and U-238 natural radioactive series respectively, however K-40 and Pb-210 were low(Table1). It was demonstrated the effectiveness of combining in-situ and in-lab radioactivity measurements to detect and characterize NORM in industrial products.

Table 1. Activity concentration C_A and expanded uncertainty $U(C_A)$, (k=2)

Radionuclide	Abrasives		Scrap		Normal Soil	
	C_A (Bqkg^{-1})	$U(C_A)$ (Bqkg^{-1})	C_A (Bqkg^{-1})	$U(C_A)$ (Bqkg^{-1})	C_A (Bqkg^{-1})	$U(C_A)$ (Bqkg^{-1})
K-40	21	±10	35	±14	299	±22
Tl-208	95,3	±6,8	97,4	±4,8	7,54	±0,64
Pb-210	25	±13	61	±16	37,3	±7,3
Pb-212	559	±36	364	±23	23,6	±1,4
Pb-214	65,2	±6,7	282	±18	24,6	±2,0
Bi-212	431	±47	329	±25	32,9	±5,7
Bi-214	36,1	±4,9	224	±11	22,9	±1,8
Ac-228	392	±27	287	±14	24,2	±2,2
Th-234	55	±18	230	±22	18,8	±5,5

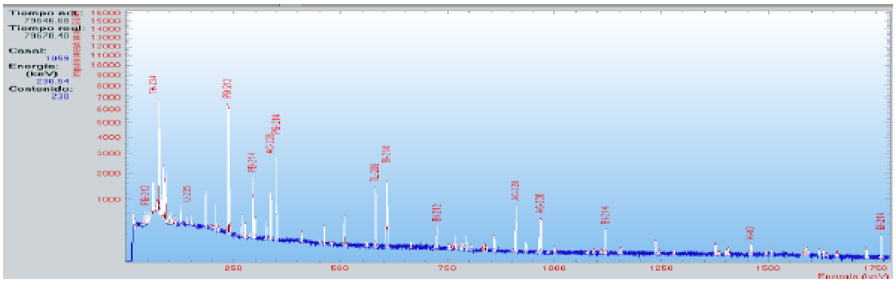

Fig. 1. Typical NORM gamma spectrum

***Keywords**: Detection, NORM, Industry*

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PS7 (T7.4-0676)

Study on the Framework of Radiation Environmental Supervision System for Radioactive Associated Minerals

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Abstract: China's radioactive associated minerals are numerous, widely distributed, involving many industries, and the level of environmental protection technology is uneven. Its radiation environment supervision started late and developed slowly. There is no uniform judgment standard for associated radioactive mineral pollution in the world. There are few things that can be directly borrowed, which are relatively weak areas. This paper summarizes the current nuclear and radiation safety laws and regulations, mineral resources development and utilization of radiation environmental safety regulations and regulations related to the associated radioactivity, points out the current shortcomings, and provides recommendations for the corresponding regulatory system research.

PS7 (T7.4-0720)

National Monitoring for Radioactivity in Food During 2012 – 2018 Period, China

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Background: Radionuclides of natural origin are routinely found in food, and radionuclides of artificial origin can also be presented and arose from authorized discharges, nuclear weapons testing or as a result of past nuclear accidents. Assessing radionuclide contamination in food is an important consideration in food safety. So, understanding the radionuclides in food and their respective activity concentrations is necessary to be able to quantify the risk of exposure for the public. Characterizing the current background of radionuclides in food makes it possible to better identify future contamination incidents and emerging long-term trends, and it could enable possible mitigation before increasing levels of contamination become a significant public health risk.

Methods: The food samples were collected and radioactivity levels were determined using HPGe gamma-ray spectrometer in 2012-2018 by local surveillance institutes. Data were obtained from National Monitoring Network for Radioactivity in Food, total number of sample in different provinces, the concentration of radionuclides in different types of food were analyzed, as a result, radiation doses due to the consumption of these foodstuffs to humans were estimated.

Results: From 2012 to 2018, nationwide monitoring for food radioactivity were carried out in China organized by NIRP. Total of 5311 samples of 7 kinds of foods commonly found in Chinese diet have been measured for 10 different radionuclides of both natural and anthropogenic origin. ^{40}K has the largest mean concentration and the highest detectable rate in all types of food. ^{238}U achieves the highest detectable rate in seaweed, and tea has the highest percent above the detection limit of ^{228}Ra, ^{226}Ra, and

PS7 (T7.4-0720)

National Monitoring for Radioactivity in Food During 2012 – 2018 Period, China

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^{137}Cs among different food categories. ^{238}U, ^{228}Ra, and ^{40}K have the highest mean concentration in tea samples, which are 2.99, 1.73 and 296 Bq/kg, wet weight, respectively. ^{226}Ra and ^{137}Cs have the highest mean concentration in milk and dairy products samples, which are 1.16 and 0.56 Bq/kg, wet weight, respectively. The annual committed effective dose of ^{238}U, ^{228}Ra, ^{226}Ra, ^{40}K, ^{137}Cs are 20.43, 109.71, 55.11, 240.4 and 0.74 μSv/a, respectively.

Conclusions: The radioactivity level of food is one of the important indicators of food safety. The national monitoring for radioactivity in food that started in year of 2012, through 7 years monitoring, ~~it is~~ proved that the monitoring network can effectively obtain the level, distribution of radioactivity in food. This work can also improve the ability of emergency monitoring of nuclear accidents. Carrying out nationwide monitoring for radioactivity in foodstuffs can detect and warn of radioactive contamination in food, present a scientific basis for decision-making in health, and provide authorities with timely information on human radiation doses.

PS7 (T7.4-0877)

Dose-rate Mapping Using Smartphone for Risk Awareness in Local Residents and Workers at Zircon Sand Facility in Bangka Island in Indonesia

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Three-quarters of Bangka Island's total area of 1.6 million hectares is used for mining, and more than 70% of the laborers work in mining-related businesses. Bangka is affluent in mineral resources such as Zircon, Monazite and Xenotime, which is radioactive since it contains Thorium and/or Uranium series. Natural background in Bangka is two to three times higher than other surrounding islands, because of such rich deposits of Uranium (total 24 kilo tons) and Thorium (total 1.5 billion tons).

This field research focuses on Zircon sand processing and storage facility as shown in Fig.1 in Bangka. Zircon is widely used in industrial products. The sand contains 0.0331% of Uranium series and 0.0148% of Thorium series. The air-dose rate was 5 to 20 uSv/h, where some intervention should be considered according to ICRP #103 [1]. A lot of sandbag clusters are located right next to ordinary homes. Since there are no entry restrictions or guidance signs, local residents including children easily enter. Workers do not have personal dosimeters and masks. Some workers hide the sandbags under their bed, since they believe it would be re-sellable like rare earth. They have limited radiation literacy, since most of them originally worked in agriculture and fishing.

In this situation, rapid survey and countermeasures are required to reduce personal exposure for the local residents and workers. We developed Android smartphone-connected, dose-rate mapping system, based on PocketGeiger [2]. Its PIN photodiode sensor is capable to detect gamma radiation (0.05 to 100uSv/h. The App logs dose rate with GPS location and capable to share the data with other measurer in standard KML and CSV which is importable for Google Map, Google Earth and other GIS applications. This paper shows a visualization of dose distribution and discuss reasonably achievable goal for radiation protection and education in local residents and workers at NORM area in developing country.

Fig. 1. Zircon Sand Storage Area in Bangka Island (5 – 20 uSv/h)

Keywords: *Citizen Science, SDGs, TENORM*

ACKNOWLEDGMENTS

Acknowledgments can be placed here if needed. (left alignment)

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PS7 (T7.4-0952)

Analysis of South Africa's Regulation of Radioactive Fracking Waste

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South Africa's current regulatory framework is inadequate to mitigate the high risks and uncertainties related with shale gas exploitation, especially when hydraulic fracturing technology is employed. The Regulations for Petroleum Exploration and Production, as published in Government Notice R466 in Government Gazette 38855, 2015 (hereafter the "Fracking Regulations"), are not strict enough and do not provide adequate checks and balances to ensure sustainable development of the shale gas resource. Wastes from shale gas extraction can contain the radioactive isotopes radium-226 (^{226}Ra) and radium-228 (^{228}Ra), which decay further into radon (Rn). Exposure to radon, a form of naturally occurring radioactive materials (NORM), has been recorded as a leading cause of lung cancer. This paper examines South Africa's current regulation on the handling the disposal of technologically enhanced naturally occurring radioactive materials (TENORM) and/or NORM waste from gas operations and compared with other countries which have been involved in regulating NORM and TENORM waste from gas operations to reduce adverse radiological health effects.

***Keywords**: Fracking, Radon, TENORM*

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PS7 (T7.4-1110)

Analysis of Natural Radionuclides Concentrations in Ceramics using Quadrupole ICP-MS

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All rocks and soils contain uranium, thorium, and their decay progeny. The distribution of these natural radionuclides in perception varies depending on the local rock formation process and may be very high locally [1]. Uranium and/or thorium compounds may be added to the glaze used on the surface of ceramics, to produce a variety of colours. Ceramics incorporating uranium compounds, arising from beta and gamma radiation emitted by uranium decay may potentially have external dose effect. In addition, ceramics is manufactured from fine soil contained alumina, silicon, iron and so on. The most commonly used clay as its raw materials in Korea is gypsum and aluminium silicate or Sancheong-soil, which is produced in Sancheong area of Gyeongsang-do, is widely used as raw material. This paper describes that the concentration of uranium and thorium from ceramics in Korea using by quadrupole inductively coupled plasma mass spectrometry (ICP-MS).

Firstly homogenous sample preparation through milling or mixing, the sample were dried in a temperature controlled oven up to 450°C for about 5hrs. For the analysis of ICP-QMS, the samples were dissolved by fusion with lithium metaborate and acid digestion with HF, HNO_3, and $HClO_4$ to break refractory complex. The dissolved solutions were precipitated at pH7 using Fe co-precipitation and dilution HNO_3. The concentration of uranium and thorium were analyzed using by ICP-QMS. In order to evaluate the natural radionuclides and compare with indirect analysis of uranium and thorium, the dried samples analyzed thorium and uranium series using by a gamma-ray spectrometer. The samples spend time for a month to attain the secular equilibrium between radium and radon isotope in decay series. Thorium and uranium series that represented as shorter-lived radionuclides Pb-214 and Bi-214 were measured using a HPGe gamma detector. The reference soil sample (SRM 2710a) with certificate or reference values were analyzed using fusion system and ICP-MS for validation results.

In general, the concentrations of uranium and thorium in soil are 16-110 Bq/kg and 11-64 Bq/kg, respectively [2]. In the case of ceramics containing uranium and/or thorium compounds as glaze, the concentration of uranium and thorium are 20-160 Bq/kg and 30-140 Bq/kg, respectively. This trend is related to the characteristics of a geological and mineral resources that was granite as the base rock.

Keywords: *NORM, ICP-MS, by-product*

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PS7 (T7.4-1145)

Total Body Potassium Survey of Radiation Workers of KAERI

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In the process of internal radio-contamination monitoring to radiation workers of KAERI in 2019, total body potassium (TBK) was additionally evaluated to confirm its level compared with other related research results on natural radiation exposure from K-40. Subjects of 352 rad-workers who are 50 women and 302 man registered in KAERI were sampled from monitoring data using In-vivo Whole Body Counter (model 2520 FASTSCAN, Canberra). Body potassium per kg of body mass of KAERI workers were 1.57 g-K for women and 1.81 g-K for man (Table 1 and Fig. 1), of which were as of 8% and 21% higher than that of other Korean research work performed by KIRAMS.[1)] Annual effective doses due to the body content of K-40 was 0.146 mSv for women and 0.169 mSv for man respectively, and these were also 67% (average of 0.167 mSv/y) higher than 0.1 mSv/y[2)] derived from the data of daily intake of foods in typical Korean foods.

Table 1. Summary of total body potassium surveys to KAERI radiation workers registered in 2019

	K-40 (Bq, s^{-1})	Subjects	Age range (y)	Weight (kg)	Height (cm)	TBK (g)	BK/kg	E (mSv)/y
Female	2,758	50	24~62	56.7 (40~85)	163 (154~172)	89	1.57	0.146
Male	4,210	302	22~69	74.8 (50~120)	173 (158~188)	136	1.81	0.169
Whole	4,026	352	22~69	72.2	172	129	1.80	0.167

BK/kg: Body Potassium in gram per weight, E: annual effective dose

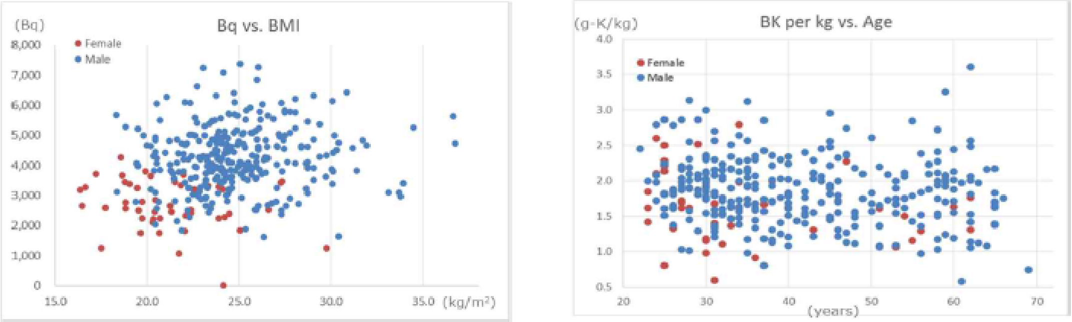

Fig.1 K-40 radioactivity with Body Mass Index (BMI, left) and body potassium concentration with age (right)

Keywords: *total body potassium, natural radioactivity of K-40, annual effective dose*

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PS7 (T7.4-1209)

Development of a Methodology for Assessing Exposure Dose in Landfill Disposal of Processed Products Containing NORM

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Interest in disposing of processed products containing Naturally Occurring Radioactive Material (NORM) has been raised with the recent issue of some processed products containing NORM in Korea. The treatment and disposal of processed products containing NORM may result in radiation exposure. Major processes that cause radiation exposure during treatment and disposal include landfill, incineration, and transport [1]. However, the methodology for radiation dose assessment of treatment and disposal for the NORM has not been clearly established in Korea. In this study, we developed the methodology for assessing exposure dose for landfill of processed products containing NORM. Exposure dose from landfill can be assessed using RESRAD-ONSITE computer program. The RESRAD-ONSITE can assess the exposure dose to workers and residents due to radioactive materials present in soil [2]. The input parameters considered in RESRAD-ONSITE include source term, calculation time, landfill information, exposure information, and ingestion information. Table 1 shows the input parameter in RESRAD-ONSITE. For source terms, the nuclides considered in this study are processed products containing NORM. Therefore, they can be set as uranium, thorium decay series, K-40 etc. The calculation time can be set for 1,000 years according to the notice of the National Nuclear Safety and Security Commission (NSSC). For landfill information, it can be evaluated by setting information corresponding to the landfill site specific data which is a factor that determines the area and thickness of the landfill site. For exposure information, which is related to the occupancy ratio, breathing rate, and shielding factor, the values that reflect domestic conditions can be used. For ingestion information, including ingestion rate, livestock consumption rate, and food contamination proportion, domestic conditions can be used. In this study, a procedure for assessing exposure dose for landfill of processed products was developed. The aforementioned variables can be used to assess the exposure dose appropriate to domestic conditions. The results of this study are expected to contribute to the establishment of a methodology for radiation dose assessment for defective processed products containing NORM in Korea.

Table 1. RESRAD-ONSITE input parameter

Input parameter	Details		
Source term	Radionuclide	Radioactive concentration	Distribution coefficient
Landfill information	Contaminated zone	Saturated zone	Unsaturated zone
Exposure information	Occupancy ratio	Breathing rate	Shielding factor
Ingestion information	Ingestion rate	Livestock consumption rate	Food contamination proportio

Keywords: NORM, Processed products, Dose assessment

ACKNOWLEDGMENTS
This work was supported through the KoFONS using the financial resource granted by NSSC. (No. 1805016)

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PS7 (T7.4-1215)

A Study of Natural Radioactivity and Chemical Composition in Cement Samples in South Korea

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Every human being has always been exposed to ionizing radiation arising from naturally occurring radionuclides (NORs) such as ^{238}U and ^{232}Th series and natural ^{40}K. Cement contains NORs, which is the main source of indoor gamma radiation exposure to human among the building construction materials (brick, sandstone, iron, rock etc.). The levels of NORs activity in cement depend on the raw material composition as well as local geological condition. Thus, the analysis of NORs present in cement can be used to assess the radiation risk of construction materials and determine the origins of NORs.

In the present study, we analyzed ten different cement samples (500 g each) collected from commercial markets in South Korea including ready-mix samples (cement and sand) used in the building constructions and general-purpose. The activity levels of NORs were measured using high purity germanium detector with the relative efficiency of 54% and graded shielding. The absolute efficiency of voluminous samples is need to be obtained for the accurate activity calculation of NORs. Recently developed ExVol at KAERI and EFFTRAN programs were used to get the absolute efficiency of voluminous samples, which were validated using radioactive certified reference materials. Based on the results of NORs activity measurements, radiological parameters, such as external hazards index (H_{ex}), gamma and alpha index ($I\gamma$), and annual effective dose (AED), were calculated for risk assessment of building materials. The activity levels were also compared with previous measurements in South Korea and other parts of the world.

Along with assessment of natural radioactivity levels, the chemical composition of cement were determined using wavelength dispersive X-ray fluorescence spectrometer (WD-XRF). The fusion glass bead sample preparation method was adopted to minimize the grain size and mineralogical effects in WD-XRF. Statistical correlation analysis between radionuclides and stable elements was carried out for source apportionment of NORs content.

Keywords: *Natural radionuclides, Radiological risk assessment, Source apportionment*

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PS7 (T7.5-0014)

Exposure to Ionizing Radiation Resulting from the Chernobyl Fallout and Childhood Cardiac Arrhythmia: A Population Based Study

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In 2005, the Institut de Radioprotection et de Sûreté Nucléaire (IRSN, France) launched a research program named EPICE (acronym for "Evaluation of Pathologies potentially Induced by CaEsium") to collect scientific information on non-cancer effects possibly induced by chronic exposures to low doses of ionizing radiation with the view of addressing a question raised by several French NGOs related to health consequences of the Chernobyl nuclear accident in children.

The implementation of the program was preceded by a pilot phase to ensure that the project would be feasible and determine the conditions for implementing an epidemiological study on a population of several thousand children. The EPICE program focused on childhood cardiac arrhythmias started in May 2009 for 4 years, in partnership with the Russian Bryansk Diagnostic Center. The purpose of this cross-sectional study was to determine the prevalence of cardiac arrhythmias in the Bryansk oblast (depending on the contamination of the territory and the caesium-137 whole-body burden) and to assess whether caesium-137 was or not a factor associated with the onset of cardiac arrhythmias. To address these questions, a study bringing together 18 152 children aged 2 to 18 years was initiated; each child received three medical examinations (ECG, echocardiography and caesium-137 whole-body activity measurement) and some of them were given with a 24-hour Holter monitoring and blood tests.

The findings of the study showed no evidence of an association between the presence of cardiac arrhythmia and caesium-137 exposure. Thus, caesium-137 is not an associated factor in the frame of the study. These results allowed us to bring clear answers to the issue of radiation-induced childhood arrhythmia, a subject that has been debated for many years. Our results will certainly useful for future comparative study in children exposed to ionizing radiation in other contexts, such as cancer radiation therapies.

PS7 (T7.5-0216)

Radiation Impact Assessment of Recycling 137Cs Polluted Soil after Nuclear Power Plant Accidents

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Abstract: After the accident at the Fukushima nuclear power plant in Japan, a large amount of radioactive contaminated soil was generated, which brought great challenges to treat and dispose the subsequent soil. The recycling of slightly polluted soil can reduce the amount of contaminated soil. This paper analyzes the main illuminated scenes of ^{137}Cs polluted soil used in the construction of highway subgrade, soil bag handling and paving. The Monte Carlo algorithm (MCNP) and RESRAD-ONSITE program were used to simulate the two scenes respectively, and the concentration of ^{137}Cs in the soil at the effective dose of 1mSv/a was derived. It is concluded that: 1) the annual effective doses of porter and paver caused by 1Bq/g ^{137}Cs contaminated soil are 0.41mSv and 0.23mSv respectively; 2) the activity of ^{137}Cs in the soil at 1mSv/a for porter and pavers are 2.44Bq/g and 4.26Bq/g respectively; 3) When recycling ^{137}Cs contaminated soil for highway construction, it is recommended that the ^{137}Cs activity concentration level be controlled below 2Bq/g.

Key words：Recycling; ^{137}Cs ;soil

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Poster Session T7

PS7 (T7.5-0294)

Comprehensive Approach to the Environmental and Public Impact Assessment at the Nuclear Legacy Sites and Surrounded Populated Areas

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The results of comprehensive environmental monitoring studies at the former coastal technical bases of the Navy in the North-West and Far Eastern regions of Russia are presented. These areas are classified as nuclear legacy sites and currently used as temporary storage facilities for spent nuclear fuel (SNF) and radioactive waste (RW) at the contaminated territories. One of the peculiarities of legacy sites is the mixed contamination of the environment with radioactive and chemical pollutants. Among the dominate dose forming artificial radionuclides ^{137}Cs and ^{90}Sr there are also ^{60}Co, ^{152}Eu, ^{154}Eu, plutonium isotopes (^{238}Pu,^{239}Pu,^{240}Pu,^{241}Pu) and ^{241}Am. Soil distribution of the radioactive contaminants within the industrial sites is characterized by the presence of locally contaminated areas resulted from radionuclide migration from the SNF and RW storage facilities which protective barriers were damaged due to unauthorized practice in the past. The contaminated soil is mainly attributed to the category of industrial and low-level radioactive waste (LLW). Non-radioactive contamination of the nuclear legacy sites is characterized by a wide range of pollutants and contamination levels. It was found strong environmental pollution (soil, surface and groundwater) with heavy metals (nickel, vanadium, lead, arsenic, beryllium, thallium, cadmium, chromium, etc.). In the international regulatory practice, the common approach is focused on the control of radiation factor at the nuclear legacy sites which are not sufficient to characterize the environment and public impact assessment during remediation of the sites. The comprehensive risk assessment approach was applied to estimate cancerogenic and non-carcinogenic risks to the population living in the vicinity of the nuclear legacy sites. It was shown that the individual carcinogenic risk from artificial radioactive contamination is negligible (1.08 E-06) while the non-carcinogenic risks from exposure to chemicals (heavy metals in drinking water) are characterized by values exceeding the acceptable risk in the range of 1.0 E-4 to 5.0 E-4.

***Keywords**: nuclear legacy, non-radiological contamination, risk assessment*

PS7 (T7.5-1171)

Changes of Absorbed Dose Rate in Air on Asphalt Paved Roads Related to Fukushima Daiichi Nuclear Power Plant Accident

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Asphalt road surfaces were contaminated with artificial radionuclides released in the Fukushima Daiichi Nuclear Power Plant (F1-NPP) accident and there presence has dramatically changed absorbed dose rate in air. The Tokatsu area (six cities) of Chiba Prefecture, Japan, located 200 km southwest of the F1-NPP was the most heavily radionuclide-contaminated area of Chiba Prefecture. Some of the present researchers made the detailed distribution map of absorbed dose rate in air in this area in 2015, and they found higher absorbed dose rates in air exceeding 80 nGy h^{-1} were observed along national highways constructed using high porosity asphalt [1]. In the present study, car-borne surveys were carried out for the whole Tokatsu area in 2020, and the changes of absorbed dose rate relating to the F1-NPP accident were observed.

The average absorbed dose rate in air for the whole Tokatsu area in 2020 was 53 nGy h^{-1}, which was 22% smaller compared to that in 2015 (68 nGyh^{-1}). The average dose rate along national highways was 60 nGy h^{-1}, which was 28% less compared to the rate measured in 2015 (83 nGyh^{-1}). Artificial radionuclides deposited on asphalt surfaces tend to be easily washed away by rainfall. However, unlike general roads, these national highways use porous asphalt that has a structure in which dust easily accumulates. Therefore, it was considered that the artificial radionuclides was electrostatically bound to this dust and formed a higher air dose rate region compared with other surrounding roads. The rate of decrease was significantly greater than physical attenuation, and even greater on national highways. The reason for this is considered to be due to weathering activities such as rainfall.

Table 1. Absorbed dose rates in air for natural and artificial radionuclides measured in 2015 and 2020.

	Absorbed dose rate in air (nGy/h)				Percent reduction (%)
	2015		2020		
	Mean±SD	Range	Mean±SD	Range	
Whole Tokatsu area (n = 2325)	68±20	201 – 25	53±11	140 – 25	22
National highways (n = 264)	83±28	176 – 30	60±15	141 – 29	28

Keywords: *Fukushima Daiichi Nuclear Power Plant accident, Absorbed dose rate in air, Radiocesium*

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PS7 (T7.5-1173)

Changes in Environmental Radiation Levels in Katsushika-ku, Tokyo after the Fukushima Daiichi Nuclear Power Plant Accident

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Artificial radionuclides were released into the environment due to the accident at the Fukushima Daiichi Nuclear Power Plant following the Great East Japan Earthquake. The released radionuclides moved through the air and were deposited on the ground over wide areas and affected by wind direction and rainfall. Katsushika-ku, Tokyo, was one place in the radioactive plume paths and high concentrations of anthropogenic radionuclides were measured [1]. We have been measuring air dose rates in Katsushika-ku, Tokyo, since 2015 to investigate the secular change of anthropogenic radionuclides. In this presentation, we report the measurement results to date.

The air dose rates were measured by the car-borne survey technique and the fixed-point measurement method (at 27 points). In the method, the counting rates were measured by driving on major roads in Tokyo. In the fixed-point measurement method, the vehicle counting rate was measured by stopping the vehicle at each point and the outside counting rate was measured. In addition, the air dose rate was calculated by unfolding the wave height distribution obtained from the outside measurements using a 22×22 row response matrix method, and the dose conversion coefficients were calculated from the correlation with the counting rate. In this study, these coefficients were used to calculate the in-vehicle counting rate to the air absorbed dose rate.

The air dose rate for the whole Katsushika-ku area obtained by the driving survey method in 2019 was 60 ± 11 Gy/h. The air dose rate for the whole Katsushika-ku area between 2015 (65 ± 14 nGy/h) and 2016 (65 ± 13 nGy/h) decreased 0.4 %, It decreased 1.9 % between 2016 and 2017 (64 ± 13 nGy/h), 3.3 % between 2017 and 2018 (62 ± 12 nGy/h) and 2.0 % between 2018 and 2019 (60 ± 11 nGy/h). The dose rate of anthropogenic radionuclides discriminated by the fixed-point measurement method was 24.9 nGy/h in 2015, but it decreased to 12.3 nGy/h in 2019, a decrease of 50.5 %, and the air dose rate of anthropogenic radionuclides in Katsushika-ku was cut to about half in the 8-year period after the accident.

Keywords: *Artificial radionuclides, Absorbed dose rate in air, car-borne survey technique*

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PS7 (7.A-0471)

Applied Ethics and Decision Making: Utilizing Decision Making Models with the Hunter's Point Naval Shipyard Superfund Site as a Case Study

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As part of the United State Environmental Protection Agency's Superfund program the Hunter's Point Naval Shipyard, located in San Francisco, California, operated as the United States Navy Radiological Defense Laboratory from 1945-1969, and has been undergoing remediation since its permanent closure in 1994. The Hunter's Point site was used for a variety of industrial and radiological activities during its operations, including, but not limited to, the study of radiation effects on animals, radiological decontamination training exercises, the decontamination and study of ships present during nuclear weapons testing from Operation Crossroads, and the production of radiological waste drums for offshore disposal. In 2014, the United States Nuclear Regulatory Commission (USNRC) opened an investigation of potential fraud by a contractor holding firm fixed price contracts for the decontamination efforts. Spurred by a letter of concern issued by the Navy Radiological Affairs Support Office (RASO) to the USNRC regarding potentially mismanaged samples and fraudulent record keeping, the investigation resulted in a Notice of Violation based on the actions of individual employees.[1] As a result of the investigation, two employees of the contractor faced criminal sentences. Recently, in January 2019, the United States Department of Justice joined three whistleblower lawsuits against the contractor, alleging that the contractor submitted false claims to the United States Navy for radiological remediation and support services.

When completing decontamination work at radiologically impacted sites, decisions regarding the extent of cleanup efforts are made regularly, and these decisions depend on the application of ethical principles. At Clemson University, ethical decision making is taught to students using models that can be applied to a wide variety of scenarios. Two models developed at Clemson University's Rutland Institute for Ethics were used to examine actions taken by staff at Hunter's Point Naval Shipyard as a case study. The first model is the Identify, Analyze, Justify, and Decide (IAJD) model, which promotes a strong emphasis on identifying stakeholders and applying tests from classical western ethics principles to consider the perspective of those stakeholders. The second model, the Stop, Test, Act, and Reflect (STAR) model, approaches the problem with a focus on testing potential solutions to ethical questions, and emphasizes reflection on one's actions as well as taking ownership of those decisions. This presentation will introduce the IAJD model and demonstrate how to apply the model to this case study; the full paper will discuss application of both decision-making models and compare their utility.

***Keywords**: Applied Ethics, Decision making models*

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PS7 (7.B-0070)

The Health Physics Society's Ask the Experts Feature: How to Impart Knowledge to a Wide Variety of People

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The Health Physics Society (HPS) website, hps.org, was created in 1999. Its original content included goals for the HPS, lists of officers and board members, information on upcoming meetings, Society rules, dues, etc. Shortly thereafter, website Editor in Chief Gen Roessler and Webmaster Fred Baes presented to the HPS Board of Directors the idea for an Ask the Experts feature on the website. The concept was to give members of the public, as well as trained health physicists (HPs), a pathway to ask questions of acknowledged experts regarding radiation protection.

The function has been a rousing success, with over 13,000 questions asked and then each personally answered by one of the nearly 300 experts who volunteer their time to craft the response. Questions have been asked in 26 categories that embody nearly 100 subcategories. The most common categories for questions are radiation effects (15.0%), medical and dental patient issues (13.0%), pregnancy and radiation (12.0%), medical and dental equipment (6.2%), nuclear medicine patient issues (6.0%), and radiation basics (5.9%). The frequency of questions about pregnancy led to creation of a short video that answers the most common questions. Another video is devoted to diagnostic x-ray exposures. Other videos are planned. An indication of the success of Ask the Experts is that the HPS website receives over 1 million visits annually.

Lessons learned during 20 years of interactions with various questioners will be detailed. Those include answering the question before giving background information, being brief while still accurate, using plain language with compassion, and relying upon and referencing scholarly materials. Some examples of questions and answers will be shown in the allotted time.

Keywords: *communication, public understanding*

PS7 (7.B-0847)

Introduction to the "BOOKLET to Provide Basic Information Regarding to Health Effects of Radiation" and Its applications

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After TEPCO's Fukushima Daiichi NPS accident, the Ministry of the Environment, Japan (MOEJ) and the National Institute of Radiological Sciences, National Institutes for Quantum and Radiological Science and Technology (QST-NIRS) collected and compiled basic knowledge on radiation, the experience of scientific experts, and the initiatives of relevant ministries and agencies concerning the health effects of radiation. With these resources, a booklet was developed to provide basic information regarding the effects of radiation on health. This booklet has been used in training sessions with a target audience of people engaged in health and medical care, welfare, and education with the aim of fostering personnel who can respond to the specific worries and health concerns of the residents of Fukushima and its neighboring prefectures.

The booklet consists of two major sections. First, a basic knowledge of radiation and scientific knowledge on the health effects of radiation are described (Chapters 1–5). Second, the environmental recovery measures, monitoring, and health management implemented in Japan after the accident have been explained (Chapters 6–10).

The booklet was translated to English in 2019. In the English version, a glossary was prepared to promote the understanding of technical terms. The booklet can be downloaded from the MOEJ's website.

http://www.env.go.jp/en/chemi/rhm/basic-info/index.html

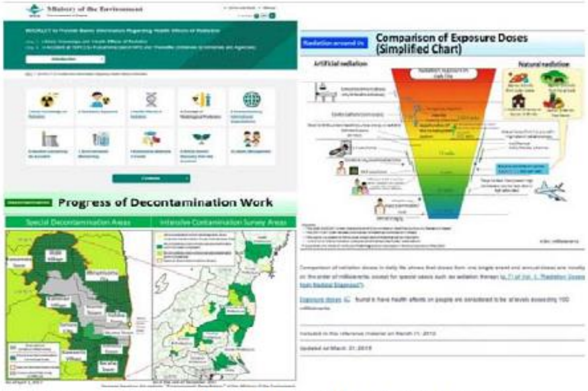

Fig. 1. Examples of the website and contents.

***Keywords**: TEPCO's Fukushima Daiichi NPS accident, radiation health effects.*

ACKNOWLEDGMENTS

We are most grateful to the group of experts who created the BOOKLET to Provide Basic Information Regarding Health Effects of Radiation.

PS8 (T8.1-1007)

Can the NIR Induced in Our Body by Stress Interact with Stress Sensitive Circuits On the Brain and Vice-Versa?

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It is well accepted that stress from life strains is dangerous to our body, the stress by itself is just how we feel and react to life strain that we are submitted daily [1, 2]. The difference between stress and strain is that stress is, for a mechanical systems, the deforming force per unit area, its unit is same as pressure, while strain is the apparent change in the shape, volume or length of object caused due to stress. The physician Hans Selye, used the term stress for the first time in the 1930s. Indeed the word "strain" had to be used rather than "stress" to avoid ambiguity [2, 3]. The changes we feel in our physical body and cognition under life pressure, is the strain caused by the stress due to work, social life, family, and internal body demands. Danger (real or not real), fear, and general social pressure triggers in our body a very complex system of electrical and biochemical processes which act as a chain reaction that are very harmful to our body, what we can call/define here as "Stress Radiation". The external non-ionizing radiation (NIR) can power the different electrical circuits in our skin and inner body such as the brain, which might be harmful to us, and so can induce stress via some bioelectrical and/or biochemical chains and the stress sensitive electric circuits on brain. To create NIR, one must have a generator – power source, an antenna to transmit the electromagnetic signal, and to receive those wave fields one must have an antenna. Part of this radiation reach us, and it is absorbed by our body causing so far not very well known biological and psychic effects. How would stress work? The HPA axis (hypothalamic–pituitary–adrenal axis) can be considered our main "antenna" to receive and process all external stress inputs. Electrochemical processes induced by stress force our body to release an ensemble of hormones such as cortisol, which is a stress hormone released by the adrenal glands. It is important for helping our body to deal with stressful situations, as our brain triggers its release in response to many different kinds of stress. However, when cortisol levels are too high for too long, this hormone can hurt us more than it helps, and this is what happens when we are under constant stress. The neurotransmitters such as Norepinephrine connect us to attentiveness, emotions, sleeping, dreaming, and learning, and it has long been known to play a prominent role in the regulation of the HPA and its modulation by glucocorticoids and stress [4]. The fight-or-flight system (the acute stress response) is a physiological reaction that occurs in response to a perceived harmful event/attack/threat to survival and so the HPA axis plays a fundamental role to connect human stressful scenarios with internal metabolically action dynamics. It is clear that our daily stress events emulate in body the attack or a threat (fight-or-flight system) - continuously, and so the cortisol production and the electrical inputs from the HPA system. Since we do not know which level of stress is acceptable for human beings, and since we do not know, yet, which level of electromagnetic field induced by stress (spurious or not) in body is safe, we suggest to consider the precaution principle when we submit anyone to stress at least at working place, which is a more controlled environment, and it might well be useful to create a safe scale for that purpose [5].

Keywords: Hans Selye, Life Strain, Brain, Stress Sensitive Circuits, Non Ionizing Radiation, Precaution Principle

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PS8 (T8.2-0730)

Assessment of Radiation Exposure to extremely Low Frequency Fields within the Vicinity of Electricity Transmission Substations in Greater Accra

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Radiation exposure to extremely low frequency (ELF) fields within the vicinity of electricity substations has been assessed and investigated. Real time magnetic field measurements at four (Volta, New Tema, Achimota and Mallam) high voltage power transmission substations (bulk supply points) have been measured using AC Milligaus Meter (Model UHS2) The maximum magnetic flux densities for occupationally exposed worker was 18.98 ± 0.15 µT, 19.77 ± 0.12 µT, 19.78 ± 0.17 µT and 19.54 ± 0.19 µT for Volta, New Tema, Achimota and Mallam substations respectively. Magnetic field flux densities measured at public access areas around the substation was 8.48 ± 0.16 µT, 19.80 ± 0.05 µT and 19.17 ± 0.11 µT for Volta, Achimota and Mallam substation respectively. The measured magnetic flux densities were above the results of similar studies of Hamza et al by a factor of about 2.5. Additionally, the results were below International Commission on Non-Ionizing Radiation Protection reference levels and also comparable with other studies reporting cancer incidence from ELF. Investigations revealed that external magnetic fields oriented perpendicularly to a human body yields maximum induced current density for all human body sizes.

Table 1. Comparisons of magnetic flux density of this study with others

Locations	Magnetic Flux Density (µT)		
	This study	Ozen	Hamza et al
Occupational occupied areas	2.66 – 19.77	0.85 – 65.10	1.00 – 8.80
Public occupied areas	8.48 – 19.80	–	–

(-) means data not available

Keywords: Extremely Low Frequency, Magnetic flux, Current density

ACKNOWLEDGMENTS
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Poster Session T8

PS8 (T8.2-1011)

A Novel Method for Preparation of Titanium Dioxide Nanoparticles for Ultraviolet Radiation Shielding

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Titanium dioxide nanoparticles (TiO_2 NPs) are widely used for ultraviolet radiation shielding. However, many conventional methods for synthesizing TiO_2 NPs have some kinds of disadvantages concerning time consumption, cost, or environmental issue. A novel technique for preparing TiO_2 NPs is developed using a one-step anodization process. When a titanium wire is anodized in 1 M potassium chloride aqueous solution at constant 15 V for 3 min at room temperature, amorphous TiO_2 NPs are produced. Various phases of TiO_2 NPs are fabricated by controlling anodization time, electrolyte composition, and applied voltage. The derived phase is the result of phase transformation due to the effects of anodization conditions. Optimizing such parameters leads to create the required phase of TiO_2 NPs. A plausible mechanism is proposed for understanding TiO_2 NPs formation. Also, the ultraviolet protection property of TiO_2 NPs fabricated by anodization is evaluated. Our TiO_2 NPs exhibit excellent ultraviolet absorbance and can be expected to be applied for ultraviolet radiation shielding, especially in sunscreens of the cosmetic field.

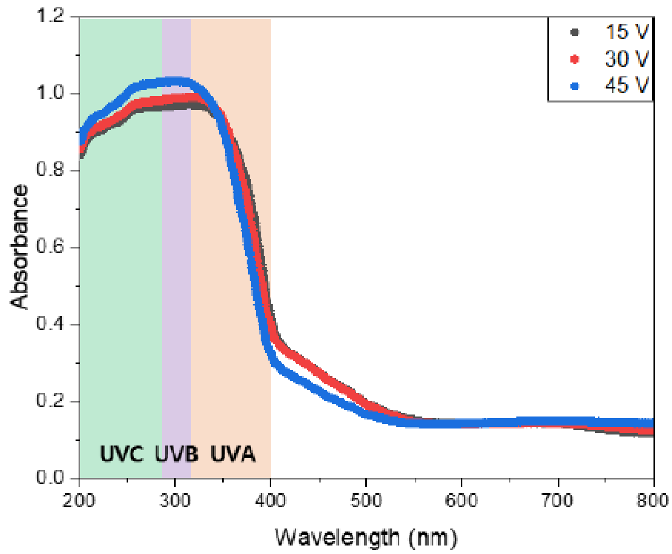

Fig. 1. UV-Vis absorption of TiO_2 NPs obtained via anodization in 1 M KCl aqueous electrolyte containing 0.05 M HCl additive at constant voltage for 15 min.

***Keywords**: Titanium dioxide nanoparticles, Anodization, Ultraviolet absorption*

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PS8 (T8.5-0834)

Localized Nonlinear Excitations in Diffusive Memristor-based Neuronal Networks

We extend the existing ordinary differential equations modeling neural electrical activity to include the memory effect of electromagnetic induction through magnetic flux, used to describe time varying electromagnetic field. Through the multi-scale expansion in the semi-discrete approximation, we show that the neural network dynamical equations can be governed by the complex Ginzburg-Landau equation. The analytical and numerical envelop soliton of this equation are reported. The results suggest from the biophysical point of view the possibility of collective information processing and transport in the nervous system, taking place in both the spatial and temporal domains in the form of a localized modulated wave. The effects of memristive synaptic electromagnetic induction coupling and perturbation on the modulated action potential dynamics examined. Large electromagnetic induction coupling strength may contribute to signal block as the amplitude of modulated waves are observed to decrease. This could help in the development of a chemical brain anaesthesia for some brain pathologies.

Key words : neural electrical activity, memory effect, electromagnetic induction

PS8 (T8.5-1126)

Electromagnetic Fields and People's Health: the Risks of Cell Phones and Antennas

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Introduction:

The cell phone has become an irreplaceable element in our lives and we must evaluate how to maintain all the benefits by reducing unnecessary risks as much as possible.

The use of cell phones has had a truly explosive development in recent years and the market continues to grow rapidly. If the current progression continues, in a couple of decades the number of cell phones would greatly exceed the number of inhabitants of the planet. In just 20 years the power density of the electromagnetic fields measured in some cities has increased much more than 100 times which means that the biological effects on exposed living things should also have been increased…

The possibility of maintaining continuous communication, without having to restrict the freedom of movement, is a quality highly appreciated by all.

However, there has been a growing concern in the public about the adverse effects caused by radiation exposure resulting from the use of this technology and the health impact such effects would have. The reality is that the sum of the information obtained in recent years of epidemiological studies on cancer, experiences with laboratory animals and in vitro studies, as well as the worrying increase in the frequency of brain tumors in countries that have good statistics such as Great Britain, France, Australia and Canada determine the urgency of applying the "Precautionary Principle" as recommended by the European Parliament, and informing the public about ways to reduce risks.

The objective of this work is to briefly describe the current state of knowledge on the subject, what are the possible risks to which the population is exposed and mention what are the most effective preventive measures to take advantage of all the benefits of new technologies and limit as much as possible the risks to which we are exposed.

Recent studies, projects and publications of greater significance:

1 - The INTERPHONE project and the new generic study:
2 - The CERENAT Project.
3 - The REFLEX Project.
4 - Cellular oxidative stress.
5 - The Danish Cohort Study.
6 - The co-carcinogenic effect.
7 - The INTEROCC Study (Co-carcinogenesis).
8 - Study of the American Toxicology Program (US-NTP).
9 - The studies of the Ramazzini Institute. NIR/IR sinergy
10 - Increased frequency of brain tumors, in particular Multiform Glioblastoma (GBM)
11 - The application of the criteria of Sir Bradford Hill to demonstrate Causality.
12 - Comparison of radioprotection criteria for NIR and IR (ICRP vs ICNIRP)
"Common sense recommendations to protect your Health and that of your Family"

Keywords: Risks, Health, Right to know

REFERENCES: In accordance with the recommendations of the IRPA Congress of La Havana (2017), CIPRACEM was established in order to stimulate the training of Experts in Radiological Protection of NIR, the dissemination of relevant scientific information in EMF and health, to avoid unjustified exposures.

PS8 (T8.5-1189)

A Feasibility Study for Standardization of Research Designs and Protocols for Safety Assessment of Extremely High-frequency Electromagnetic Fields

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As research and development of advanced wireless systems progress, new wireless utilization is expected to expand in the near future, increasing opportunities for human exposure to radio waves of newly utilized frequencies. Therefore, it is important to accumulate research data that will contribute to health risk assessments of possible adverse effects by exposure to newly utilized frequencies. Then, we launched a research project called "STandardization of Experimental Protocol for Safety assessment of EMF (STEPS EMF)." In this study, a feasible concept of test protocol applying for evaluation of the effect of extremely high-frequency electromagnetic field used for the 5th generation of telecommunication network was studied.

Electromagnetic waves in the 28 GHz band have a wavelength of about 10 mm and are well absorbed by water. Therefore, as a characteristic, in the case of exposure to the human body from next-generation communication devices such as 5G, it was considered that the human body surface would absorb the electromagnetic wave. Also, when absorbed the electromagnetic wave, it is mainly converted into thermal energy, but its penetration depth is estimated to be approximately 300 to 500 μm at 28 GHz. Thus, in evaluating the safety of humans to the electromagnetic wave in the environment, it was considered appropriate to target tissues on the body surface where electromagnetic wave is absorbed, that is, skin and cornea. Based on this, considering the test systems such as the OECD test guidelines[1)], the surface of the reconstructed human epidermis model used in "In vitro skin irritation: Reconstructed human epidermis test method (TG439)" is basically suitable for *in vitro* protocol, because it is a kind of target tissues and possible to directly expose the electromagnetic wave. Evaluation of characteristics of human reconstructed tissue models are underway. On the other hand, an exposure system for this purpose should be operated double-blind manner. Then, we designed an exposure system to use double lines of identical exposure device that is randomly assigned and controlled by a computer without any notice for experiment operator.

In addition, a part of global survey to collect the opinion of researchers about EMF issue and to assess the possibility of the global consensus regarding standardization of research protocol. Questionnaire was developed through the discussion by this project group. For instance, questionnaire was divided to three parts; 1) general information of responders, 2) common questions for all responders, 3) questions in each specialized research area (*in vitro*, *in vivo*, and engineering). Summary of response will present at the conference.

***Keywords**: extremely high-frequency electromagnetic field, fifth generation of mobile telecommunication, health risk assessment*

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